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Bound Report



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SPENT FUEL STORAGE EXPANSION

at

ST. LUCIE UNITS 1 AND 2

for

FLORIDA POWER & LIGHT

HOLTEC PROJECT NO. 1201

HOLTEC REPORT HI-2022882

REPORT CATEGORY: A

REPORT CLASS: SAFETY RELATED

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SUMMARY OF REVISIONS

Revision 1 contains the following pages:	
COVER PAGE	1 page
DOCUMENT ISSUANCE AND REVISION STATUS	1 page
SUMMARY OF REVISIONS	1 page
TABLE OF CONTENTS	9 pages
1.0 INTRODUCTION	6 pages
2.0 CASK PIT STORAGE RACKS	27 pages
3.0 MATERIAL AND HEAVY LOADS CONSIDERATIONS	16 pages
4.0 CRITICALITY SAFETY ANALYSES	36 pages
-- APPENDIX 4A	26 pages
5.0 THERMAL-HYDRAULIC CONSIDERATIONS	36 pages
6.0 STRUCTURAL/SEISMIC CONSIDERATIONS	61 pages
7.0 FUEL HANDLING AND CONSTRUCTION ACCIDENTS	16 pages
8.0 FUEL POOL STRUCTURE INTEGRITY CONSIDERATIONS	30 pages
9.0 RADIOLOGICAL EVALUATION	4 pages
10.0 INSTALLATION	7 pages
11.0 ENVIRONMENTAL COST/BENEFIT ASSESSMENT	7 pages
TOTAL	284 pages

Revision 1

Client editorial comments, provided in fax dated August 28, 2002, required changes to the following pages; 2-3, 3-10, 3-12, 4-9, 4-10, 4-24, 4-26, 4-27, 4-31, 5-1, 5-3, 5-4, 5-5, 5-12, 5-13, 5-15 thru 5-16, 5-18, and 5-22 thru 5-25. Faxed client comments are saved in Holtec network file G:\projects\1201\client. The following pages were revised to reflect the most recent summary results; 8-7, 8-8, and 8-18. The following pages were revised to denote proprietary information; 3-15, 4-4, 4-6, 4-7, 4-8, 4-28, 4-34, 4-36, 5-6, 5-7, 6-23, 6-24, 6-29, 7-3, and 7-4.

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St Lucie Plant is located on Hutchinson Island in St. Lucie County, Florida, south of the city of Fort Pierce. The plant consists of two Combustion Engineering Pressurized Water Reactor (PWR) nuclear units. Unit 1 has been in commercial operation since 1976 and Unit 2 since 1983.

St. Lucie Unit 1 is projected to lose full core reserve (FCR) in its Spent Fuel Pool (SFP) following Cycle 19, which ends in 2005. St. Lucie Unit 2 is projected to lose full core reserve in its Spent Fuel Pool following Cycle 17, which ends in 2007. Florida Power and Light intends to expand spent fuel storage capacity by adding a new rack within the Cask Pit of each Unit. This modification would increase the licensed storage capacity in Unit 1 from the current 1,706 storage cells to 1,849 storage cells and in Unit 2 from the current 1,360 storage cells to 1,585 storage cells. This report provides the design basis, analysis methodology, and results for the proposed spent fuel storage racks at St. Lucie to support the licensing process.

The storage expansion will add one 11 by 13 (143 total) cell Region 1 style storage rack to the Unit 1 Cask Pit and one 15 by 15 (225 total) cell Region 2 style storage rack to the Unit 2 Cask Pit. The physical descriptions of 'Region 1' and 'Region 2' racks are provided in Section 2 of this report. The functional fuel storage capabilities and safety margins are discussed in Section 4 of this report. The proposed fuel storage rack arrays for Units 1 and 2 are shown in the plan views provided by Figures 1.1.1 and 1.1.2, respectively.

The new Cask Pit storage racks are freestanding and self-supporting. The principal construction materials for the SFP racks are SA240-Type 304L stainless steel sheet and plate stock, and SA564-630 (precipitation hardened stainless steel) for the adjustable support spindles. The only non-stainless material utilized in the rack is the neutron absorber material, which is a boron carbide and aluminum-composite sandwich available under the patented product name Boral™.

The racks are designed to the stress limits of, and analyzed in accordance with, Section III, Division 1, Subsection NF of the ASME Boiler and Pressure Vessel (B&PV) Code [1]. The material procurement,

analysis, fabrication, and installation of the rack modules conform to 10CFR50 Appendix B requirements.

The rack design and analysis methodologies employed are a direct evolution of previous license applications. This report documents the design and analyses performed to demonstrate that the racks meet all governing requirements of the applicable codes and standards, in particular, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", USNRC (1978) and 1979 Addendum thereto [2].

Sections 2 and 3 of this report provide an abstract of the design and material information on the racks.

Section 4 provides a summary of the methods and results of the criticality evaluations performed for the new and spent fuel storage racks. The criticality safety analysis requires that the neutron multiplication factor for the stored fuel array be bounded by the USNRC k_{eff} limit of 0.95 under assumptions of 95% probability and 95% confidence. The criticality safety analysis sets the requirements on the Boral panel length and the amount of B^{10} per unit area (i.e., loading density) of the Boral panel for the new racks.

Thermal-hydraulic consideration requires that fuel cladding will not fail due to excessive thermal stress, and that the steady state pool bulk temperature will remain within the 150°F limit prescribed in the UFSAR to satisfy the pool structural strength, operational, and regulatory requirements. The thermal-hydraulic analyses carried out in support of this storage expansion effort are described in Section 5.

Rack module structural analysis requires that the primary stresses in the rack module structure will remain below the ASME B&PV Code (Subsection NF) [1] allowables. Demonstrations of seismic and structural adequacy are presented in Section 6.0. The structural qualification also requires that the subcriticality of the stored fuel will be maintained under all postulated accident scenarios. The structural consequences of these postulated accidents are evaluated and presented in Section 7 of this report.

Section 8 discusses the evaluation of the Cask Pit structure to withstand the new rack loads. The radiological considerations are documented in Section 9.0. Sections 10, and 11, respectively, discuss the

salient considerations in the installation of the new racks, and a cost/benefit and environmental assessment to establish the superiority of the wet storage expansion option.

All computer programs utilized to perform the analyses documented in this report are benchmarked and verified. These programs have been utilized by Holtec International in numerous license applications over the past decade.

The analyses presented herein clearly demonstrate that the rack module arrays possess wide margins of safety in respect to all considerations of safety specified in the OT Position Paper [2], namely, nuclear subcriticality, thermal-hydraulic safety, seismic and structural adequacy, radiological compliance, and mechanical integrity.

1.1 References

- [1] American Society of Mechanical Engineers (ASME), Boiler & Pressure Vessel Code, Section III, 1989 Edition, Subsection NF, and Appendices.
- [2] USNRC, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, April 14, 1978, and Addendum dated January 18, 1979.

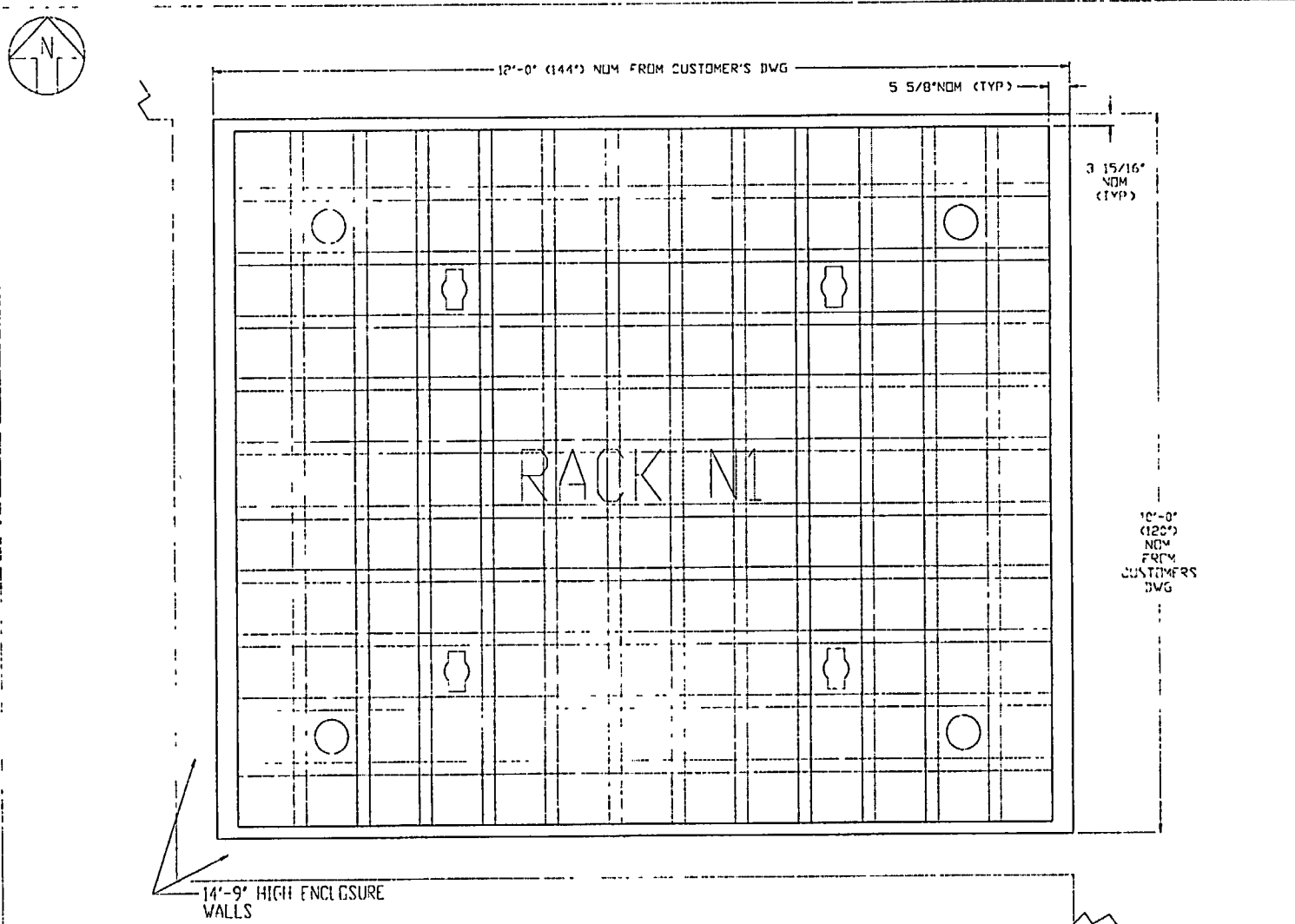
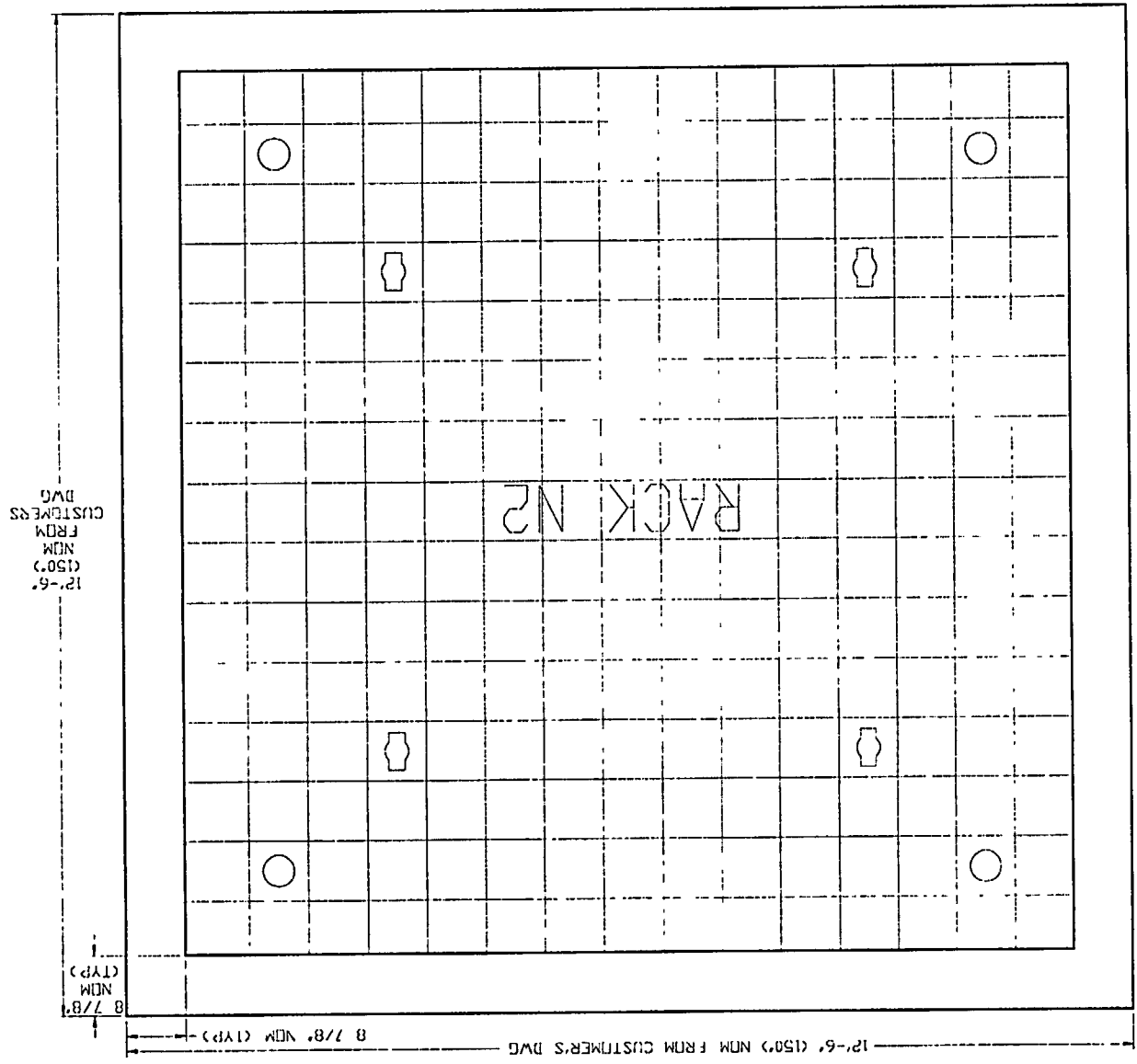


FIGURE 1.1.1; UNIT 1 CASK PIT RACK LAYOUT

FIGURE 11.2, UNIT 2 CASK PIT RACK LAYOUT



2.0 CASK PIT STORAGE RACKS

2.1 Introduction

The Unit 1 St. Lucie Cask Pit fuel storage rack will be an 11 x 13 Region 1 storage rack with a storage capacity of 143 assemblies. The Unit 2 St. Lucie Cask Pit fuel storage rack will be a 15 x 15 Region 2 storage rack with a storage capacity of 225 assemblies. Each rack will be a freestanding module, made primarily from Type 304L austenitic stainless steel containing honeycomb storage cells interconnected through longitudinal welds. Boral cermet panels containing a high areal loading of the boron-10 (B-10) isotope provide appropriate neutron attenuation between adjacent storage cells.

Figures 2.1.1 and 2.1.2 provide isometric schematics of typical Region 1 and Region 2 storage rack modules, respectively. Data on the cross sectional dimensions, weight and cell count for each rack module is presented in Table 2.1.1. The spent fuel rack modules that do not utilize flux traps between storage cells are referred to as Region 2 style racks in wet storage technology terminology. Region 1 style racks contain a water gap (a.k.a flux trap) between storage cells to provide greater margin against reactivity, thereby allowing more reactive fuel to be stored within.

The baseplates on all spent fuel rack modules extend out beyond the rack module periphery wall such that the plate protrusions act to center the rack in the pit, and establish a required minimum separation between the rack and the walls. Each spent fuel rack module is supported by four pedestals, which are remotely height-adjustable. Thus, the racks can be made plumb and the top of the racks can easily be made co-planar with the racks in the adjacent pool. The rack module support pedestals are engineered to accommodate minor level adjustments.

The elevation of the Cask Pit floor liner in each Unit is approximately three feet lower than the floor liner in the adjacent spent fuel pool. Between the rack module pedestals and the Cask Pit floor liner is a rack platform, which serves to lift the new rack to the SFP rack height, and to diffuse the dead load of the loaded racks into the reinforced concrete structure of the pit slab. The height of the platform is designed to adjust the top of rack height to the same level as the racks in the adjacent spent fuel pool. A schematic view of one of these platforms is shown in Figure 2.1.3.

The overall design of the rack modules is similar to those presently in service in the spent fuel pools at many other nuclear plants, among them Davis-Besse, Callaway, and Byron-Braidwood. Altogether, over 50 thousand storage cells of this design have been provided by Holtec International to various nuclear plants around the world.

2.2 Summary of Principal Design Criteria

The key design criteria for the new Cask Pit racks are set forth in the USNRC memorandum entitled "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", dated April 14, 1978 as modified by amendment dated January 18, 1979. The individual sections of this report address the specific design bases derived from the above-mentioned "OT Position Paper". The design bases for the new racks are summarized in the following:

- a. Disposition: Both new rack modules are required to be free-standing.
- b. Kinematic Stability: Each freestanding module must be kinematically stable (against tipping or overturning) if a seismic event is imposed.
- c. Structural Compliance: All primary stresses in the rack modules must satisfy the limits postulated in Section III subsection NF of the ASME B & PV Code.
- d. Thermal-Hydraulic Compliance: The spatial average bulk pool temperature is required to remain below 150°F in the wake of a normal partial core offload (with a single failure) or a full core offload (without a single failure).
- e. Criticality Compliance: The Unit 1, Region 1 rack must be able to store fresh Zircaloy clad fuel of 4.50 ± 0.05 weight percent (w/o) maximum enrichment while maintaining the reactivity (K_{eff}) less than 0.95. The Unit 2, Region 2 rack must be able to store Zircaloy

clad fuel of 4.50 ± 0.05 w/o maximum enrichment with a minimum burnup of 36,000 MWD/MTU while maintaining the reactivity (K_{eff}) less than 0.95.

- f. Accident Events: In the event of postulated drop events (uncontrolled lowering of a fuel assembly, for instance), it is necessary to demonstrate that the subcritical geometry of the rack structure is not compromised.

The foregoing design bases are further articulated in Sections 4 through 7 of this licensing report.

2.3 Applicable Codes and Standards

The following codes, standards and practices are used as applicable for the design, construction, and assembly of the fuel storage racks. Additional specific references related to detailed analyses are given in each section.

a. Design Codes

- (1) American Institute of Steel Construction (AISC) Manual of Steel Construction, 9th Edition, 1989.
- (2) American National Standards Institute/ American Nuclear Society ANSI/ANS-57.2-1983, "Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants" (contains guidelines for fuel rack design).
- (3) ASME B & PV Code Section III, 1989 Edition; ASME Section IX, 1989 Edition.
- (4) American Society for Nondestructive Testing SNT-TC-1A, June 1980, Recommended Practice for Personnel Qualifications and Certification in Non-destructive Testing.
- (5) American Concrete Institute Building Code Requirements for Reinforced Concrete (ACI 318-71).
- (6) Code Requirements for Nuclear Safety Related Concrete Structures, ACI 349-76/ACI 349R-76, and ACI 349.1R-80.
- (7) ASME Y14.5M, Dimensioning and Tolerancing
- (8) ASME B & PV Code, Section II-Parts A and C, 1989 Edition.

- (9) ASME B & PV Code NCA3800 - Metallic Material Organization's Quality System Program.

b. Standards of American Society for Testing and Materials (ASTM)

- (1) ASTM E165 - Standard Test Method for Liquid Penetrant Examination.
- (2) ASTM A240 - Standard Specification for Heat-Resisting Chromium and Chromium-Nickel Stainless Steel Plate, Sheet and Strip for Pressure Vessels.
- (3) ASTM A262 - Standard Practices for Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steel.
- (4) ASTM A276 - Standard Specification for Stainless Steel Bars and Shapes.
- (5) ASTM A479 - Standard Specification for Stainless Steel Bars and Shapes for use in Boilers and other Pressure Vessels.
- (6) ASTM A564 - Standard Specification for Hot-Rolled and Cold-Finished Age-Hardening Stainless Steel Bars and Shapes.
- (7) ASTM C750 - Standard Specification for Nuclear-Grade Boron Carbide Powder.
- (8) ASTM A380 - Standard Practice for Cleaning, Descaling, and Passivation of Stainless Steel Parts, Equipment and Systems.
- (9) ASTM C992 - Standard Specification for Boron-Based Neutron Absorbing Material Systems for Use in Nuclear Spent Fuel Storage Racks.
- (10) ASTM E3 - Standard Practice for Preparation of Metallographic Specimens.
- (11) ASTM E190 - Standard Test Method for Guided Bend Test for Ductility of Welds.

c. Welding Code:

ASME B & PV Code, Section IX - Welding and Brazing Qualifications, 1989.

d. Quality Assurance, Cleanliness, Packaging, Shipping, Receiving, Storage, and Handling

- (1) ANSI N45.2.1 - Cleaning of Fluid Systems and Associated Components during Construction Phase of Nuclear Power Plants - 1973 (R.G. 1.37).
- (2) ANSI N45.2.2 - Packaging, Shipping, Receiving, Storage and Handling of Items for Nuclear Power Plants - 1972 (R.G. 1.38).
- (3) ANSI N45.2.6 - Qualifications of Inspection, Examination, and Testing Personnel for the Construction Phase of Nuclear Power Plants - 1978. (R.G. 1.58).
- (4) ANSI N45.2.8 - Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Plants - 1975 (R.G. 1.116).
- (5) ANSI N45.2.11 - Quality Assurance Requirements for the Design of Nuclear Power Plants - 1974 (R.G. 1.64).
- (6) ANSI N45.2.12 - Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants - 1977 (R.G. 1.144).
- (7) ANSI N45.2.13 - Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants - 1976 (R. G. 1.123).
- (8) ANSI N45.2.23 - Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants - 1978 (R.G. 1.146).
- (9) ASME B & PV Code, Section V, Nondestructive Examination, 1992 Edition.
- (10) ANSI N16.9-75 - Validation of Calculation Methods for Nuclear Criticality Safety.
- (11) ASME NQA-1 – Quality Assurance Program Requirements for Nuclear Facilities.
- (12) ASME NQA-2 – Quality Assurance Requirements for Nuclear Power Plants.

e. USNRC Documents

- (1) "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978, and the modifications to this document of January 18, 1979.
- (2) NUREG 0612, "Control of Heavy Loads at Nuclear Power Plants", USNRC, Washington, D.C., July, 1980.

f. Other ANSI Standards (not listed in the preceding)

- (1) ANSI/ANS 8.1 - Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors.
- (2) ANSI/ANS 8.17 - Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors.
- (3) ANSI N45.2 - Quality Assurance Program Requirements for Nuclear Power Plants - 1977.
- (4) ANSI N45.2.9 - Requirements for Collection, Storage and Maintenance of Quality Assurance Records for Nuclear Power Plants - 1974.
- (5) ANSI N45.2.10 - Quality Assurance Terms and Definitions - 1973.
- (6) ANSI N14.6 - American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 pounds (4500 kg) or more for Nuclear Materials - 1993.
- (7) ANSI/ASME N626-3 - Qualification and Duties of Specialized Professional Engineers.
- (8) ANSI/ANS- 57.3 – Design Requirements for New Fuel Storage Facilities at Light Water Reactor Plants.

g. Code-of-Federal Regulations (CFR)

- (1) 10CFR20 - Standards for Protection Against Radiation.
- (2) 10CFR21 - Reporting of Defects and Non-compliance.
- (3) 10CFR50 Appendix A - General Design Criteria for Nuclear Power Plants.
- (4) 10CFR50 Appendix B - Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants.

- (5) 10CFR61 - Licensing Requirements for Land Disposal of Radioactive Waste.
- (6) 10CFR71 - Packaging and Transportation of Radioactive Material.
- (7) 10CFR100 – Reactor Site Criteria

h. Regulatory Guides (RG)

- (1) RG 1.13 - Spent Fuel Storage Facility Design Basis (Revision 2 Proposed).
- (2) RG 1.25 - Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors, Rev. 0 - March, 1972.
- (3) RG 1.28 - Quality Assurance Program Requirements - Design and Construction, Rev. 2 - February, 1979 (endorses ANSI N45.2).
- (4) RG 1.33 – Quality Assurance Program Requirements.
- (5) RG 1.29 - Seismic Design Classification, Rev. 2 - February, 1976.
- (6) RG 1.31 - Control of Ferrite Content in Stainless Steel Weld Metal.
- (7) RG 1.38 - Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water-Cooled Nuclear Power Plants, Rev. 2 - May, 1977 (endorses ANSI N45.2.2).
- (8) RG 1.44 - Control of the Use of Sensitized Stainless Steel.
- (9) RG 1.58 - Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel, Rev. 1 - September 1980 (endorses ANSI N45.2.6).
- (10) RG 1.60 – Design Response Spectra for Seismic Design of Nuclear Power Plants.
- (11) RG 1.61 - Damping Values for Seismic Design of Nuclear Power Plants, Rev. 0, 1973.
- (12) RG 1.64 - Quality Assurance Requirements for the Design of Nuclear Power Plants, Rev. 2 - June, 1976 (endorses ANSI N45.2.11).
- (13) RG 1.71 - Welder Qualifications for Areas of Limited Accessibility.
- (14) RG 1.74 - Quality Assurance Terms and Definitions, Rev. 2 - February, 1974 (endorses ANSI N45.2.10).

- (15) RG 1.85 - Materials Code Case Acceptability - ASME Section III, Division 1.
- (16) RG 1.88 - Collection, Storage and Maintenance of Nuclear Power Plant Quality Assurance Records, Rev. 2 - October, 1976 (endorses ANSI N45.2.9).
- (17) RG 1.92 - Combining Modal Responses and Spatial Components in Seismic Response Analysis, Rev. 1 - February, 1976.
- (18) RG 1.116 - Quality Assurance Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems, Rev. 0-R - May, 1977 (endorses ANSI N45.2.8-1975)
- (19) RG 1.123 - Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants, Rev. 1 - July, 1977 (endorses ANSI N45.2.13).
- (20) RG 1.124 - Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports, Revision 1, January, 1978.
- (21) RG 1.144 - Auditing of Quality Assurance Programs for Nuclear Power Plants, Rev. 1 - September, 1980 (endorses ANSI N45.2.12-1977)
- (22) RG 3.4 - Nuclear Criticality Safety in Operations with Fissionable Materials at Fuels and Materials Facilities.
- (23) RG 8.8 - Information Relative to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as Reasonably Achievable (ALARA).
- (24) IE Information Notice 83-29 - Fuel Binding Caused by Fuel Rack Deformation.
- (25) RG 8.38 - Control of Access to High and Very High Radiation Areas in Nuclear Power Plants, June, 1993.

i. Branch Technical Position

- (1) CPB 9.1-1 - Criticality in Fuel Storage Facilities.

j. American Welding Society (AWS) Standards

- (1) AWS D1.1 - Structural Welding Code - Steel.
- (2) AWS D1.3 - Structure Welding Code - Sheet Steel.
- (3) AWS D9.1 - Sheet Metal Welding Code.

- (4) AWS A2.4 - Standard Symbols for Welding, Brazing and Nondestructive Examination.
- (5) AWS A3.0 - Standard Welding Terms and Definitions.
- (6) AWS A5.12 - Specification for Tungsten and Tungsten Alloy Electrodes for Arc-Welding and Cutting
- (7) AWS QC1 - Standard for AWS Certification of Welding Inspectors.
- (8) AWS 5.4 – Specification for Stainless Steel Electrodes for Shielded Metal Arc Welding.
- (9) AWS 5.9 – Specification for Bare Stainless Steel Welding Electrodes and Rods.

2.4 Quality Assurance Program

The governing quality assurance requirements for design and fabrication of the spent fuel racks are stated in 10CFR50 Appendix B. Holtec's Nuclear Quality Assurance program complies with this regulation and is designed to provide a system for the design, analysis and licensing of customized components in accordance with various codes, specifications, and regulatory requirements.

The manufacturing of the racks will be carried out by Holtec's designated manufacturer, U.S. Tool & Die, Inc. (UST&D). The Quality Assurance system enforced on the manufacturer's shop floor shall provide for all controls necessary to fulfill all quality assurance requirements. UST&D has manufactured high-density racks for over 60 nuclear plants around the world. UST&D has been audited by the nuclear industry group Nuclear Procurement Issues Committee (NUPIC), and the Quality Assurance branch of the USNRC Office of Nuclear Material Safety and Safeguards (NMSS) with satisfactory results.

The Quality Assurance System that will be used by Holtec to install the racks is also controlled by the Holtec Nuclear Quality Assurance Manual and by the St. Lucie site-specific requirements.

2.5 Mechanical Design

The St. Lucie rack modules are designed as cellular structures such that each fuel assembly has a square opening with conforming lateral support and a flat horizontal-bearing surface. All of the storage locations are constructed with multiple cooling flow holes to ensure that redundant flow paths for the coolant are available. The basic characteristics of the Cask Pit racks are summarized in Tables 2.5.1 and 2.5.2.

A central objective in the design of the new rack modules is to maximize structural strength while minimizing inertial mass and dynamic response. Accordingly, the rack modules have been designed to simulate multi-flange beam structures resulting in excellent de-tuning characteristics with respect to the applicable seismic events. The next subsection presents an item-by-item description of the rack modules in the context of the fabrication methodology.

2.6 Rack Fabrication

The object of this section is to provide a brief description of the rack module construction activities, which enable an independent appraisal of the adequacy of design. The pertinent methods used in manufacturing the Cask Pit racks may be stated as follows:

1. The rack modules are fabricated in such a manner that the storage cell surfaces, which would come in contact with the fuel assembly, will be free of harmful chemicals and projections (e.g., weld splatter).
2. The component connection sequence and welding processes are selected to reduce fabrication distortions.
3. The fabrication process involves operational sequences that permit immediate accessibility for verification by the inspection staff.

4. The racks are fabricated per the UST&D Appendix B Quality Assurance program, which ensures, and documents, that the fabricated rack modules meet all of the requirements of the design and fabrication documents.
5. The storage cells are connected to each other by austenitic stainless steel corner welds which leads to a honeycomb lattice construction. The extent of welding is selected to "detune" the racks from the seismic input motion

2.6.1 Rack Module for Region 1

This section describes the constituent elements of the St. Lucie Unit 1, Region 1 rack modules in the fabrication sequence. Figure 2.1.1 provides a schematic view of a typical Region 1 rack.

The rack module manufacturing begins with fabrication of the "box". The boxes are fabricated from two precision formed channels by seam welding in a machine equipped with copper chill bars and pneumatic clamps to minimize distortion due to welding heat input. Figure 2.6.1 shows the box. The minimum weld seam penetration is 80% of the box metal gage, which is 0.075 inch (14 gage).

A die is used to flare out one end of the box to provide the tapered lead-in (Figure 2.6.2). One-inch diameter holes are punched on at least two sides near the other end of the box to provide the requisite auxiliary flow holes.

Each box constitutes a storage location. Each external box side is equipped with a stainless steel sheathing, which holds one integral Boral sheet (poison material), except the boxes on the rack periphery, which only have Boral on the interior sides. The design objective calls for attaching Boral tightly on the box surface. This is accomplished by die forming the internal and external box sheathings, as shown in Figure 2.6.3. The flanges of the sheathing are attached to the box using skip welds and spot welds. The sheathings serve to locate and position the poison sheet accurately, and to preclude its movement under seismic conditions.

Having fabricated the required number of composite box assemblies, they are joined together in a fixture using connector elements in the manner shown in Figure 2.6.4. Figure 2.6.5 shows an elevation view of two storage cells of a Region 1 rack module. A representative connector element is also shown in the figure. Joining the cells by the connector elements results in a well-defined shear flow path, and essentially makes the box assemblage into a multi-flanged beam-type structure. The "baseplate" is

attached to the bottom edge of the boxes. The baseplate is a 0.75 inch thick austenitic stainless steel plate stock which has 5-1/4 inch diameter holes (except lift locations, which are rectangular) cut out in a pitch identical to the box pitch. The baseplate is attached to the cell assemblage by fillet welding the box edge to the plate.

In the final step, adjustable leg supports (shown in Figure 2.6.6) are welded to the underside of the baseplate. The adjustable legs provide a $\pm 1/2$ -inch vertical height adjustment at each leg location.

Appropriate NDE (nondestructive examination) occurs on all welds including visual examination of sheathing welds, box longitudinal seam welds, box-to-baseplate welds, and box-to-box connection welds; and liquid penetrant examination of support leg welds, in accordance with the design drawings.

2.6.2 Rack Module for Region 2

Region 2 storage cell locations have a single poison panel between adjacent cell boxes on the wall surfaces separating them. The significant components (discussed below) of the Unit 2, Region 2 rack are: (1) the storage box subassembly (2) the baseplate, (3) the neutron absorber material, (4) the sheathing, and (5) the support legs.

1. Storage cell box subassembly: As described for Region 1, the boxes are fabricated from two precision formed channels by seam welding in a machine equipped with copper chill bars and pneumatic clamps to minimize distortion due to welding heat input. Figure 2.6.1 shows the box.

Each box has two lateral holes punched near its bottom edge to provide auxiliary flow holes. As shown in Figure 2.6.3, sheathing is attached to each side of the box with the poison material installed in the sheathing cavity. The edges of the sheathing and the box are welded together to form a smooth edge. The box, with integrally connected sheathing, is referred to as the "composite box".

The composite boxes are arranged in a checkerboard array to form an assemblage of storage cell locations (Figure 2.6.7). Filler panels and corner angles are welded to the edges of boxes at the outside boundary of the rack to make the peripheral formed cells. The inter-box welding and pitch adjustment are accomplished by small longitudinal connectors. This assemblage of box assemblies is welded edge-to-edge as shown in Figure 2.6.7, resulting in a honeycomb structure with axial, flexural and torsional rigidity depending on the extent of intercell welding provided. It can be seen from Figure 2.6.7 that two edges of each interior box are connected to the contiguous boxes resulting in a well-defined path for "shear flow".

2. Baseplate: The baseplate provides a continuous horizontal surface for supporting the fuel assemblies. The baseplate has a 5-1/4 inch diameter hole (except lift locations which are

rectangular) in each cell location as described in the preceding section. The baseplate is attached to the cell assemblage by fillet welds.

3. The neutron absorber material: As mentioned in the preceding section, Boral is used as the neutron absorber material.
4. Sheathing: As described earlier, the sheathing serves as the locator and retainer of the poison material.
5. Support legs: As stated earlier, all support legs are the adjustable type (Figure 2.6.6). The top (female threaded) portion is made of austenitic steel material. The bottom part is made of 17:4 Ph series stainless steel to avoid galling problems.

Each support leg is equipped with a readily accessible socket to enable remote leveling of the rack after its placement in the pool.

An elevation view of three contiguous Region 2 cells is shown in Figure 2.6.8.

TABLE 2.1.1: GEOMETRIC AND PHYSICAL DATA FOR CASK PIT RACKS								
PSL Unit	MODULE I.D.	RACK TYPE	NO. OF CELLS		MODULE ENVELOPE SIZE		WEIGHT (lbs)	NO. OF CELLS PER RACK
			N-S Direction	E-W Direction	N-S	E-W		
2	N2	Region 2	15	15	132.305"	132.305"	29,054	225
1	N1	Region 1	11	13	112.105"	132.705"	34,200	143

Table 2.5.1	
MODULE DATA FOR UNIT 1 REGION 1 CASK PIT RACK †	
Storage cell inside nominal dimension	8.55in.
Cell pitch	10.3in.
Storage cell height (above the plate)	180.0 in.
Baseplate hole size (except for lift location)	5.25 in.
Baseplate thickness	0.75 in.
Support pedestal height	4.25 in.
Support pedestal type	Remotely adjustable pedestals
Number of support pedestals per rack	4
Number of cell walls containing 1" diameter flow holes at base of cell wall	All Four Cell Walls
Remote lifting and handling provisions	Yes
Poison material	Boral
Poison length	140 in.
Poison width	7.25 in.

† All dimensions indicate nominal values

Table 2.5.2	
MODULE DATA FOR UNIT 2 REGION 2 CASK PIT RACK †	
Storage cell inside nominal dimension	8.58 in.
Cell pitch	8.80 in.
Storage cell height (above the plate)	180.0 in.
Baseplate hole size (except for lift location)	5.25 in.
Baseplate thickness	0.75 in.
Support pedestal height	4.25 in.
Support pedestal type	Remotely adjustable pedestals
Number of support pedestals per rack	4
Minimum number of cell walls containing 1" diameter supplemental flow holes at base of each cell located away from pedestals	2
Number of cell walls containing 1" diameter flow holes at base of each cell located above a pedestal	4
Remote lifting and handling provisions	Yes
Poison material	Boral
Poison length	140 in.
Poison width	7.25 in.

† All dimensions indicate nominal values

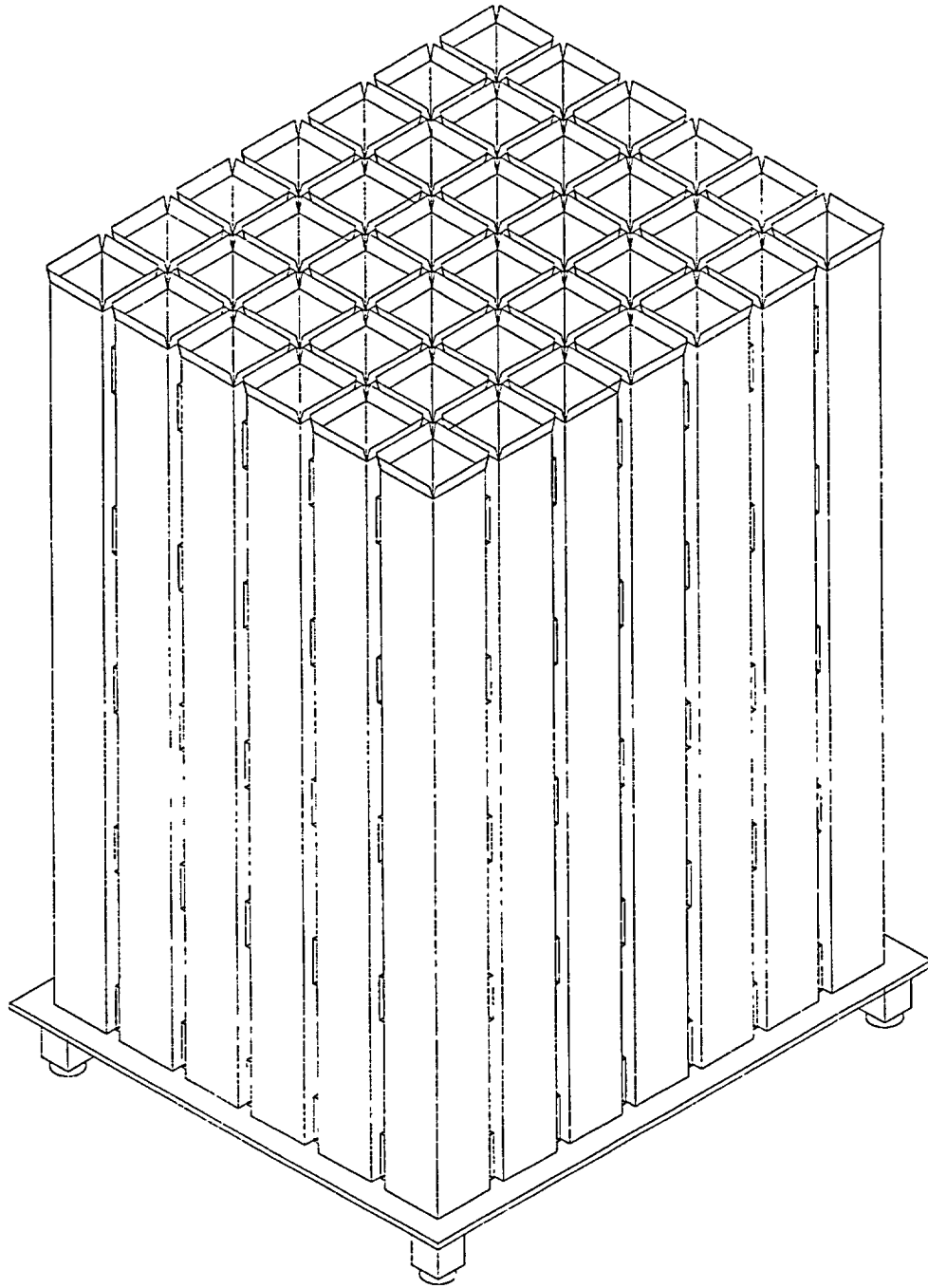


FIGURE 2.11) SCHEMATIC VIEW OF TYPICAL REGION 1 RACK STRUCTURE

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G:\DRAWINGS\12C\REL\PKT-HE-2022882\FIG211

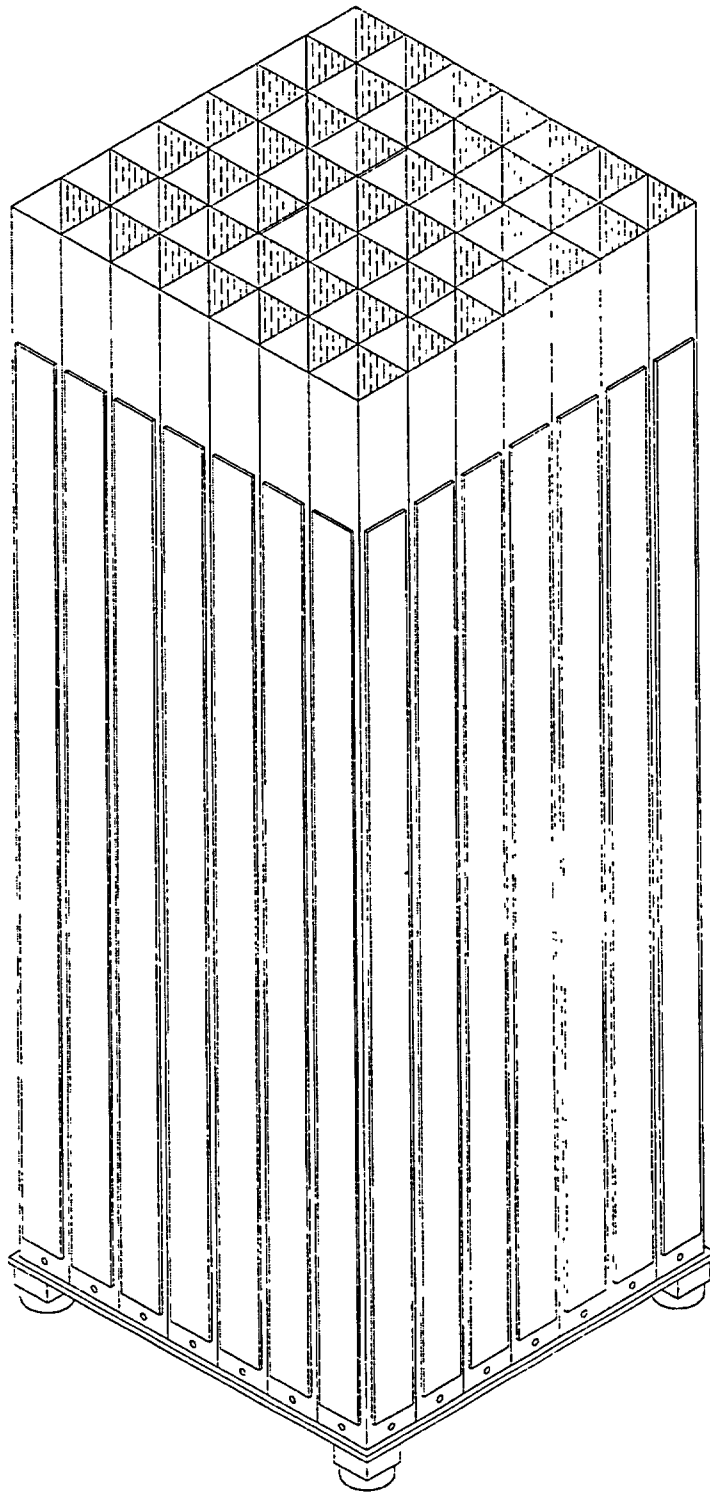


FIGURE 2.1.2; SCHEMATIC OF TYPICAL REGION 2 RACK STRUCTURE

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DRAWINGS NOT FIGURE 2.1.2 PART HI-2022882 FIGURE 2.1.2

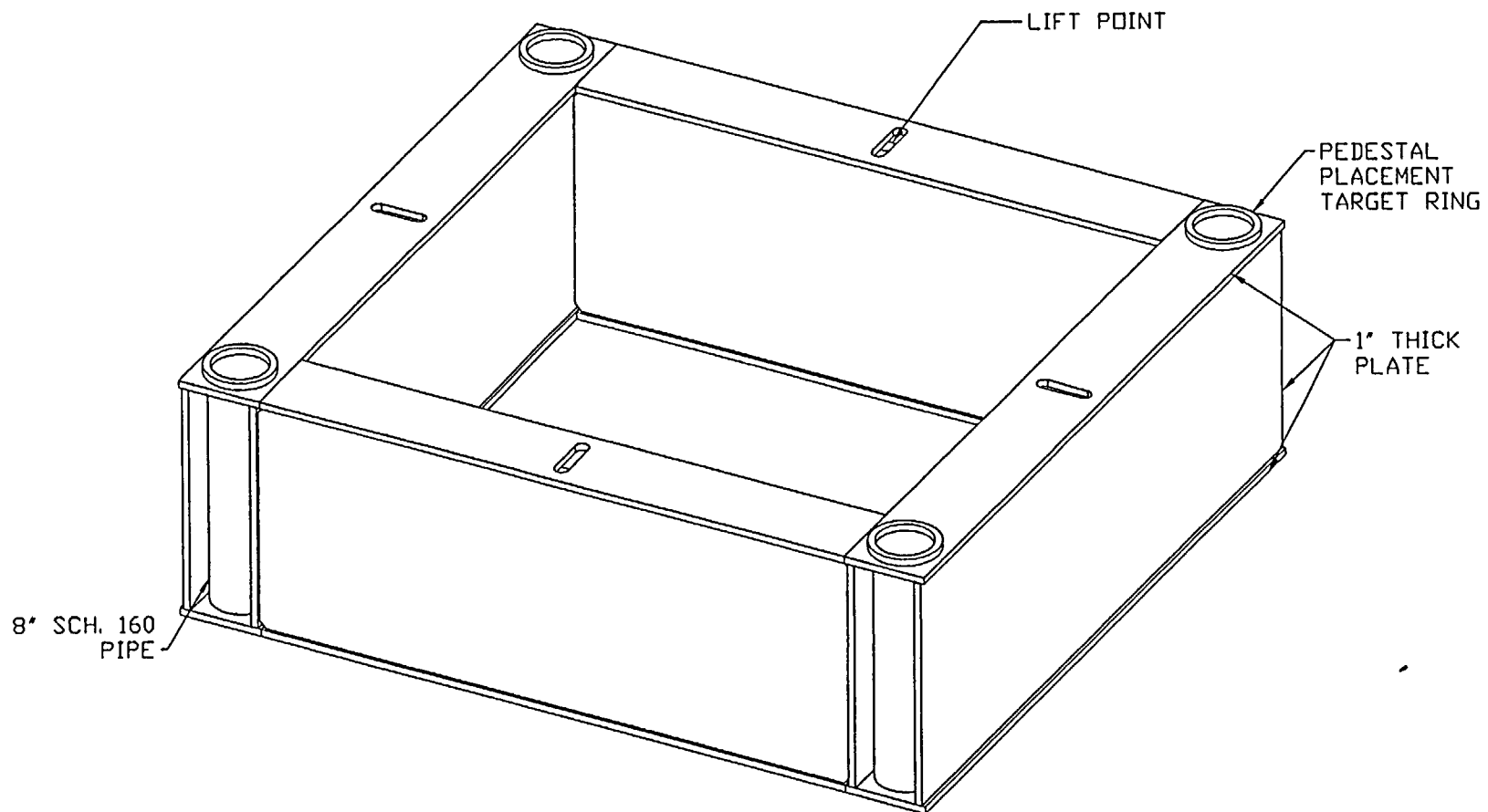


FIGURE 2.1.3;
CASK PIT RACK PLATFORM

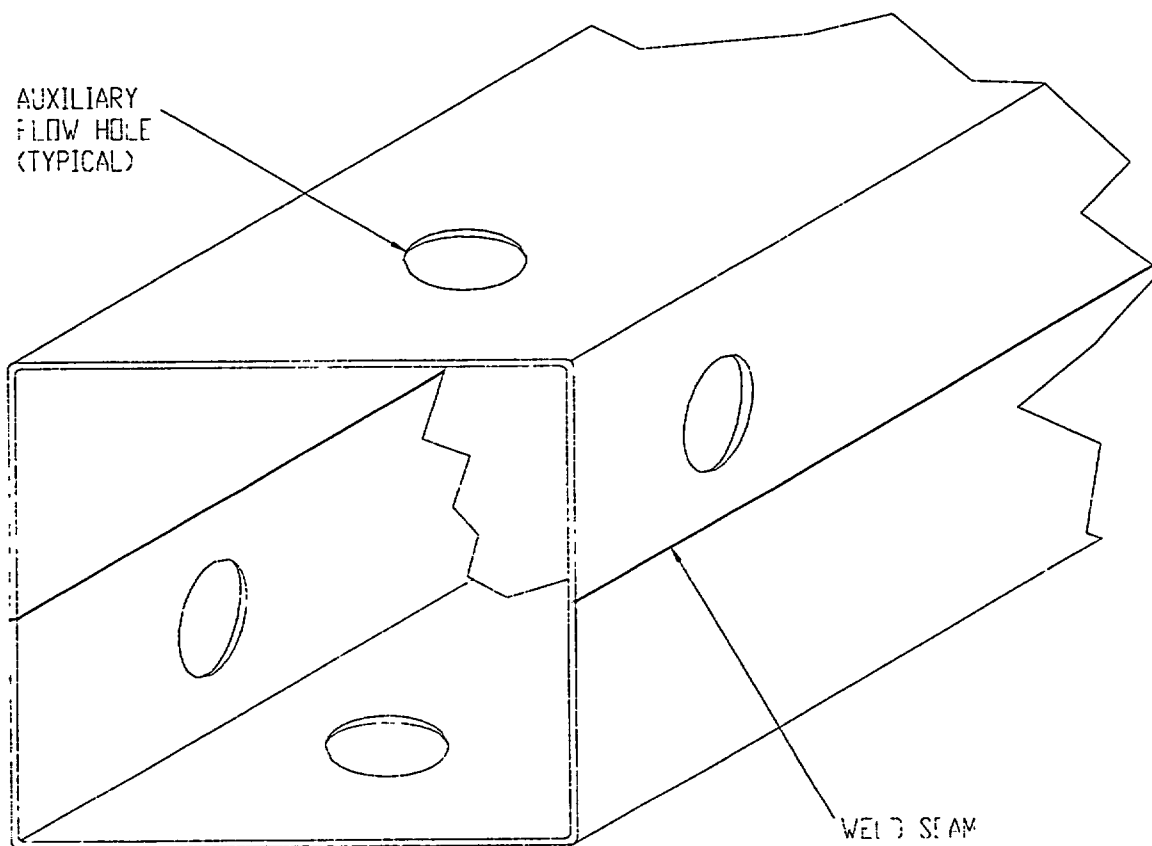


FIGURE 2.6.1; SEAM WELDED PRECISION FORMED CHANNELS

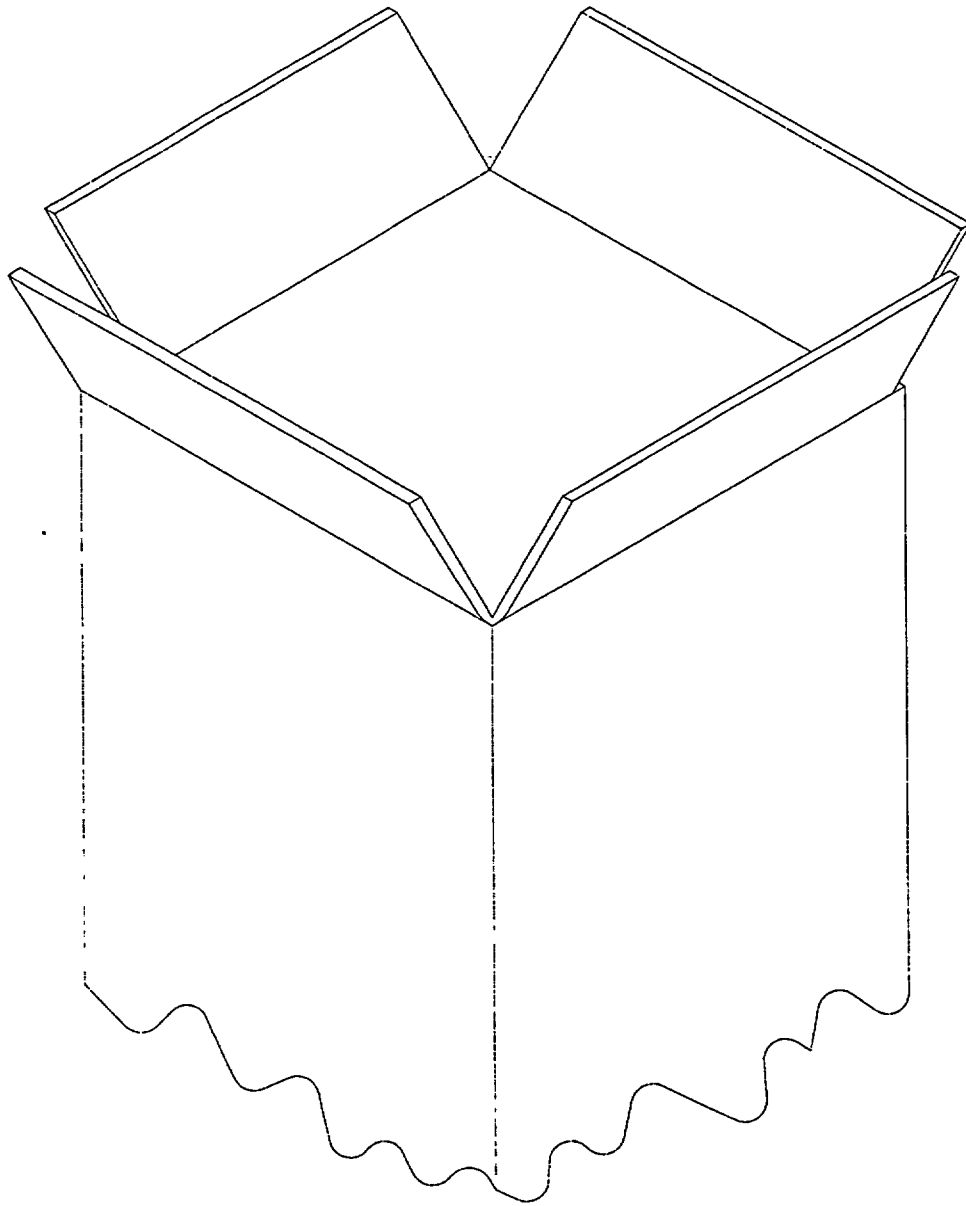


FIGURE 2.6.2; TAPERED REGION 1
CELL END

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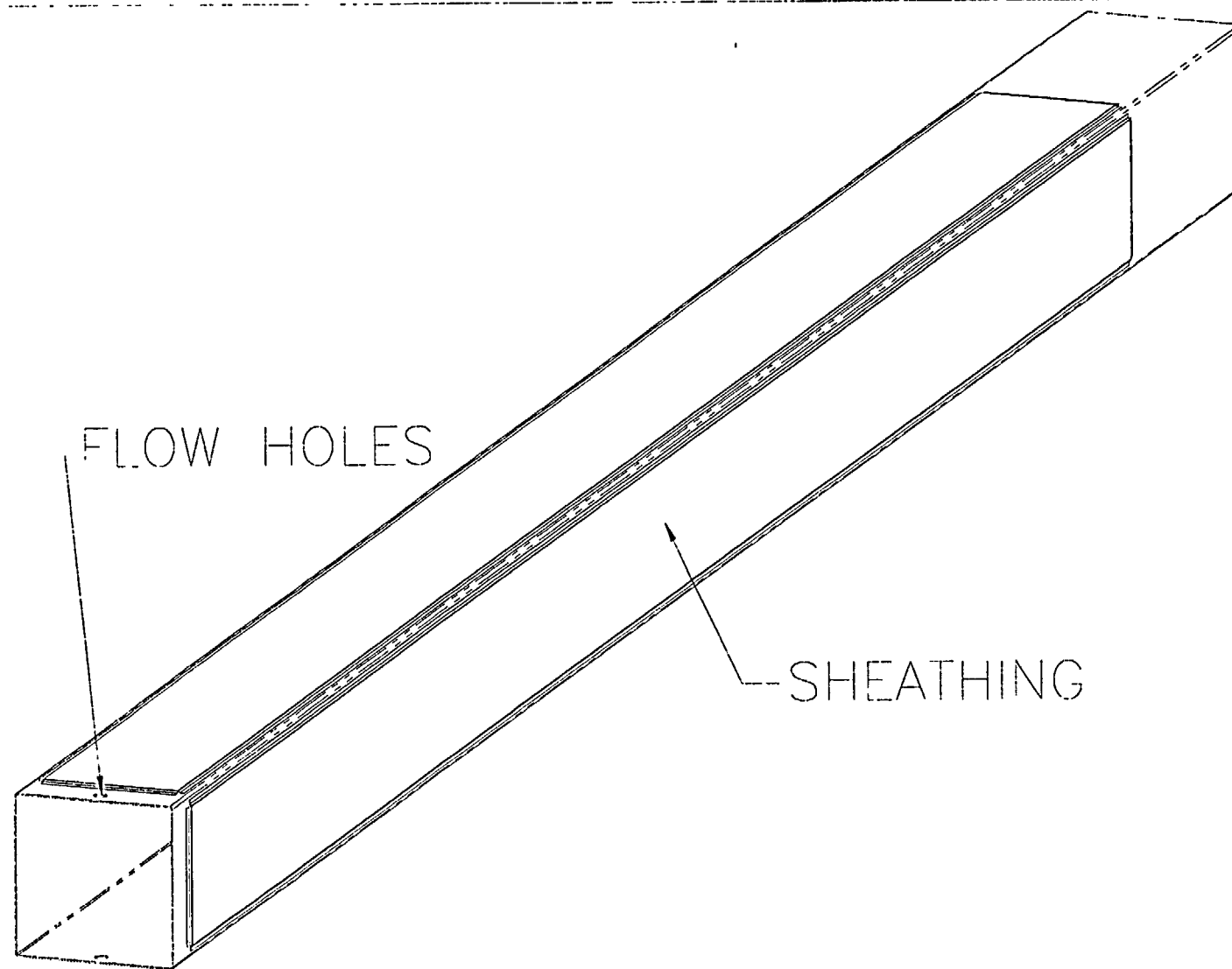


FIGURE 2.6.3; COMPOSITE BOX ASSEMBLY

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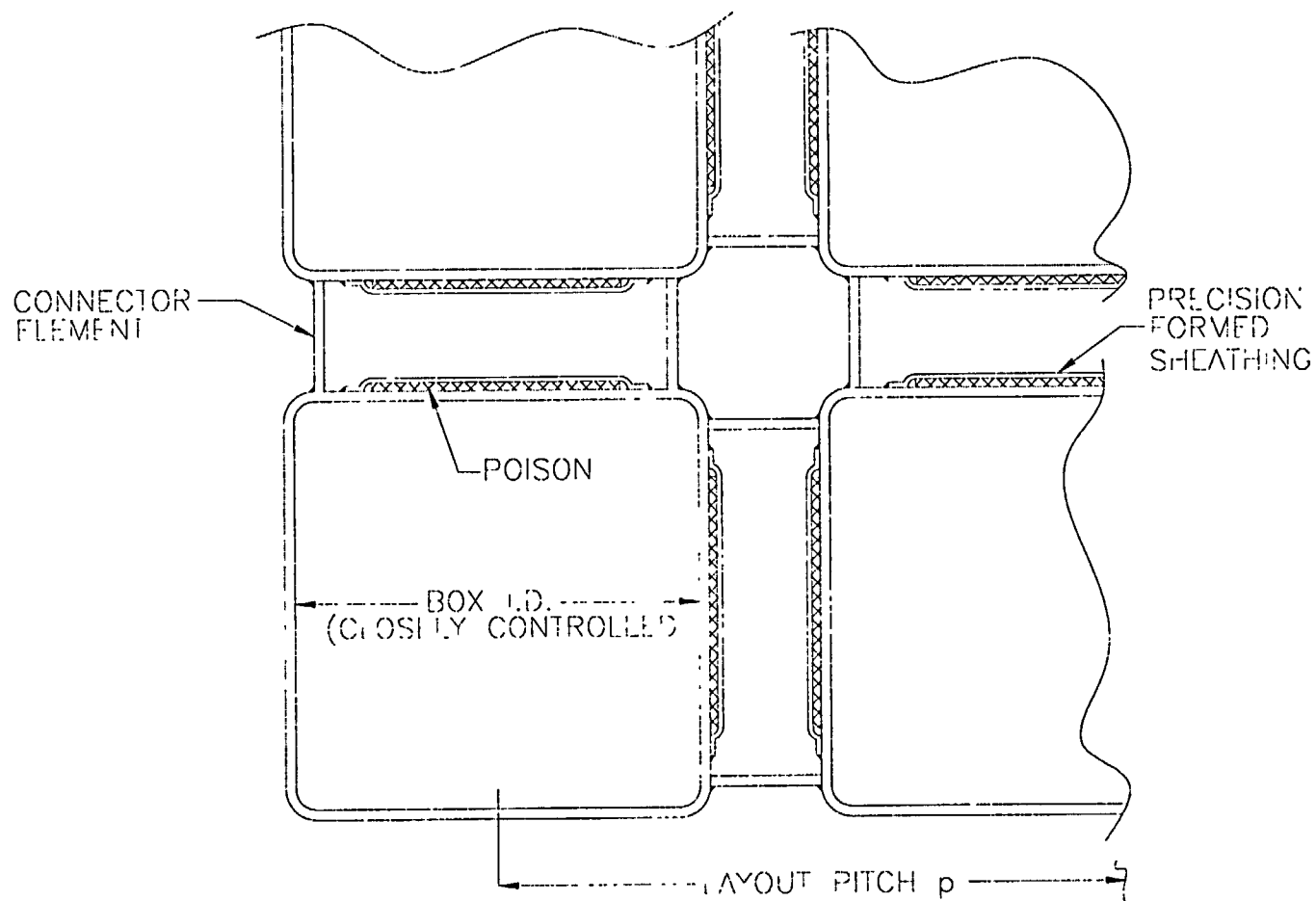


FIGURE 2.6.4; ASSEMBLAGE OF REGION 1 CELLS

H1201

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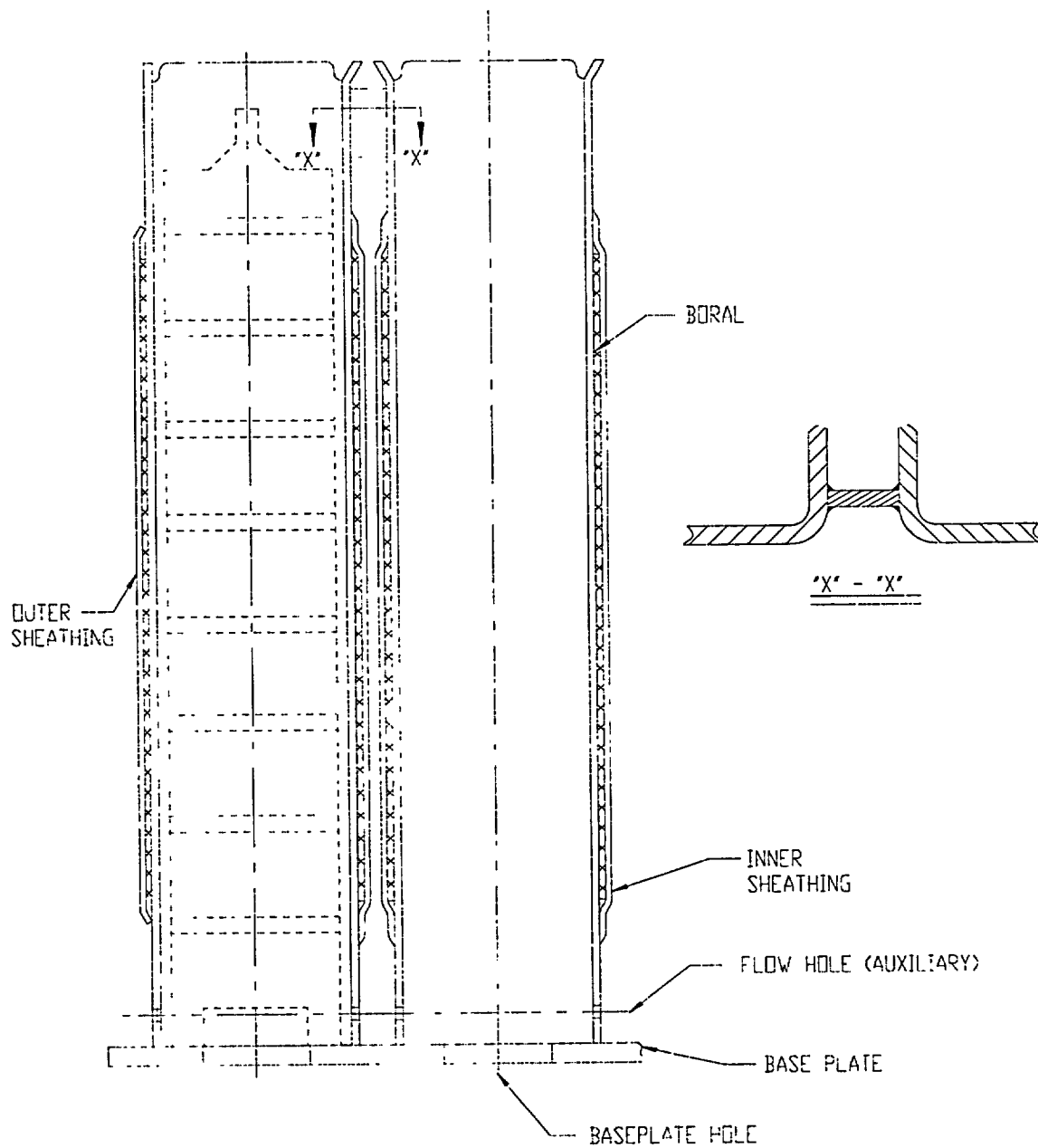


FIGURE 2.6.5: ELEVATION VIEW OF REGION 1 RACK

NOTE: DEPICTION OF STORED FUEL ASSEMBLY
IS NOT INTENDED TO BE ACCURATE

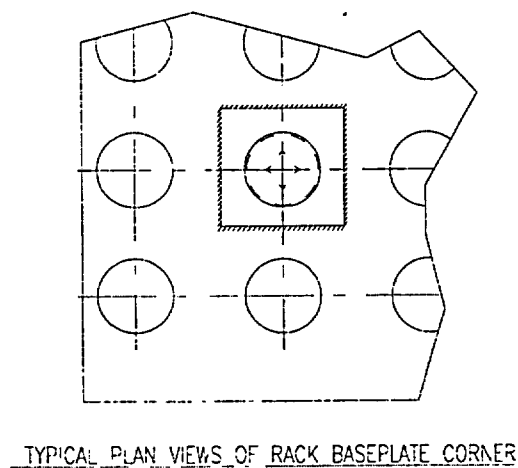
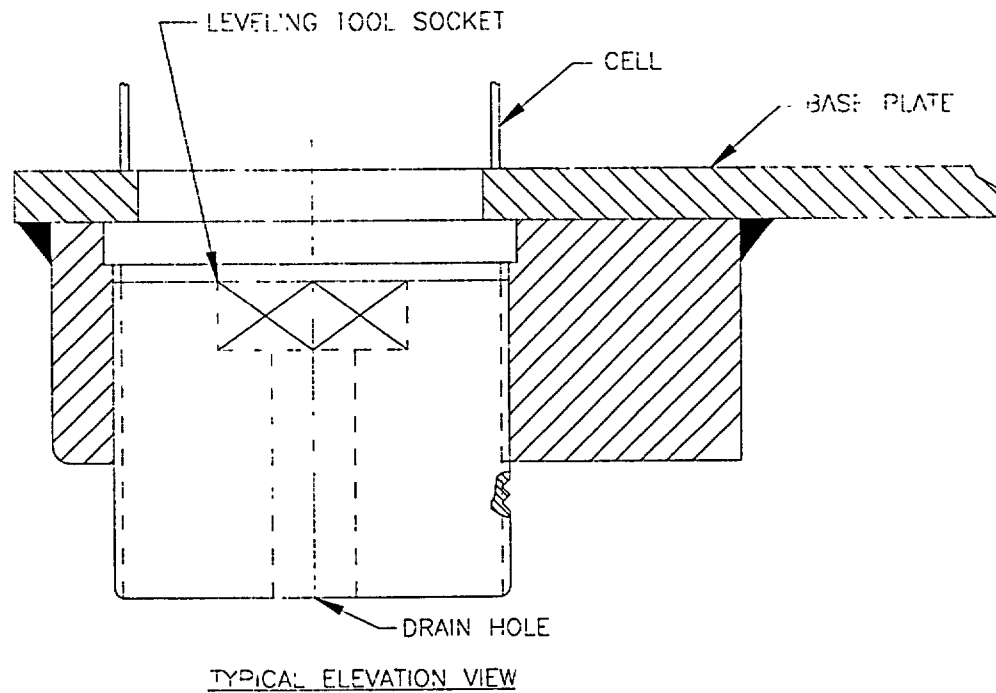


FIGURE 2.6.6; SUPPORT PEDESTALS FOR HOLTEC
PWR RACKS

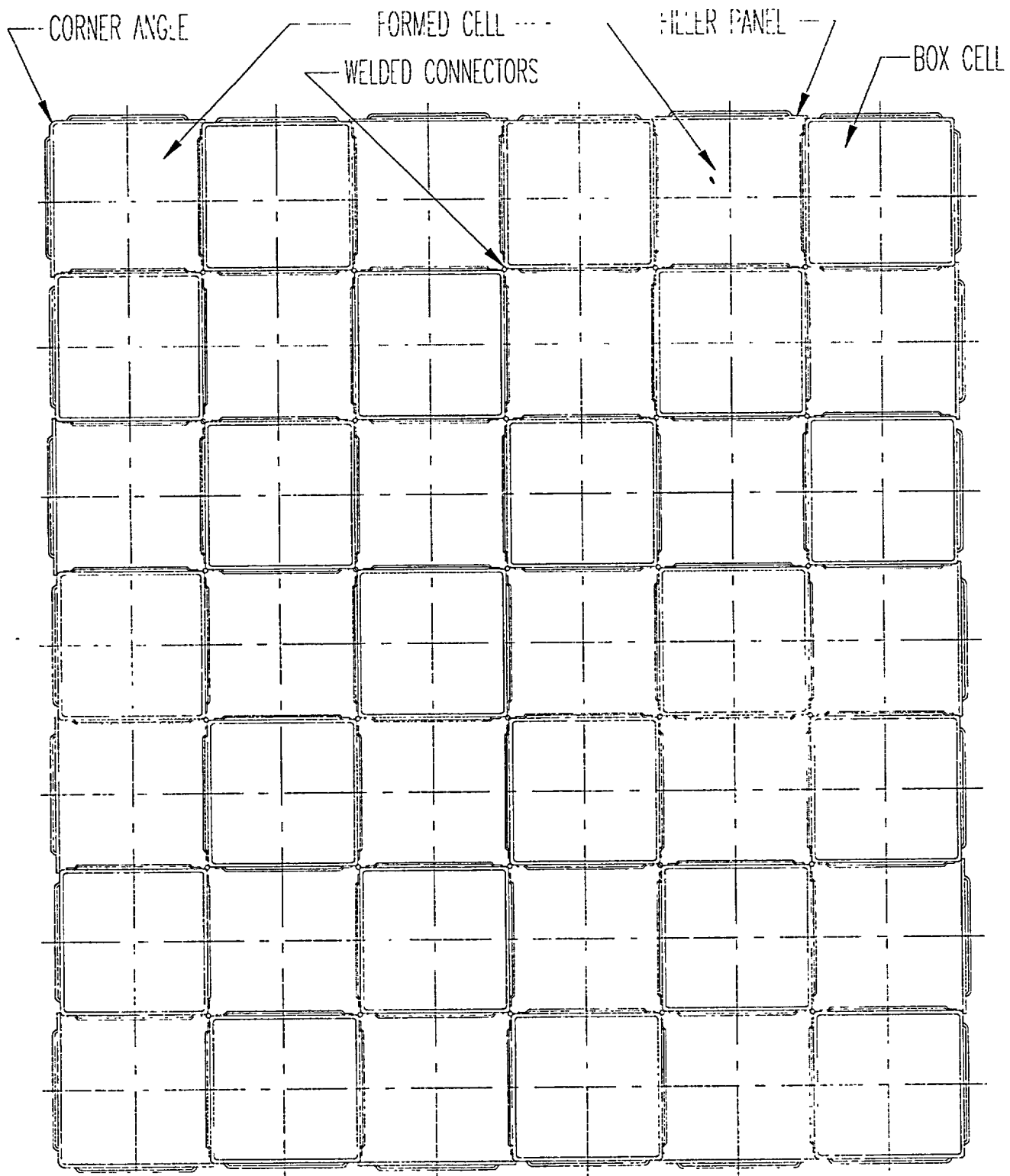


FIGURE 2.6.7; TYPICAL ARRAY OF STORAGE CELLS
(NON-FLUX TRAP CONSTRUCTION)

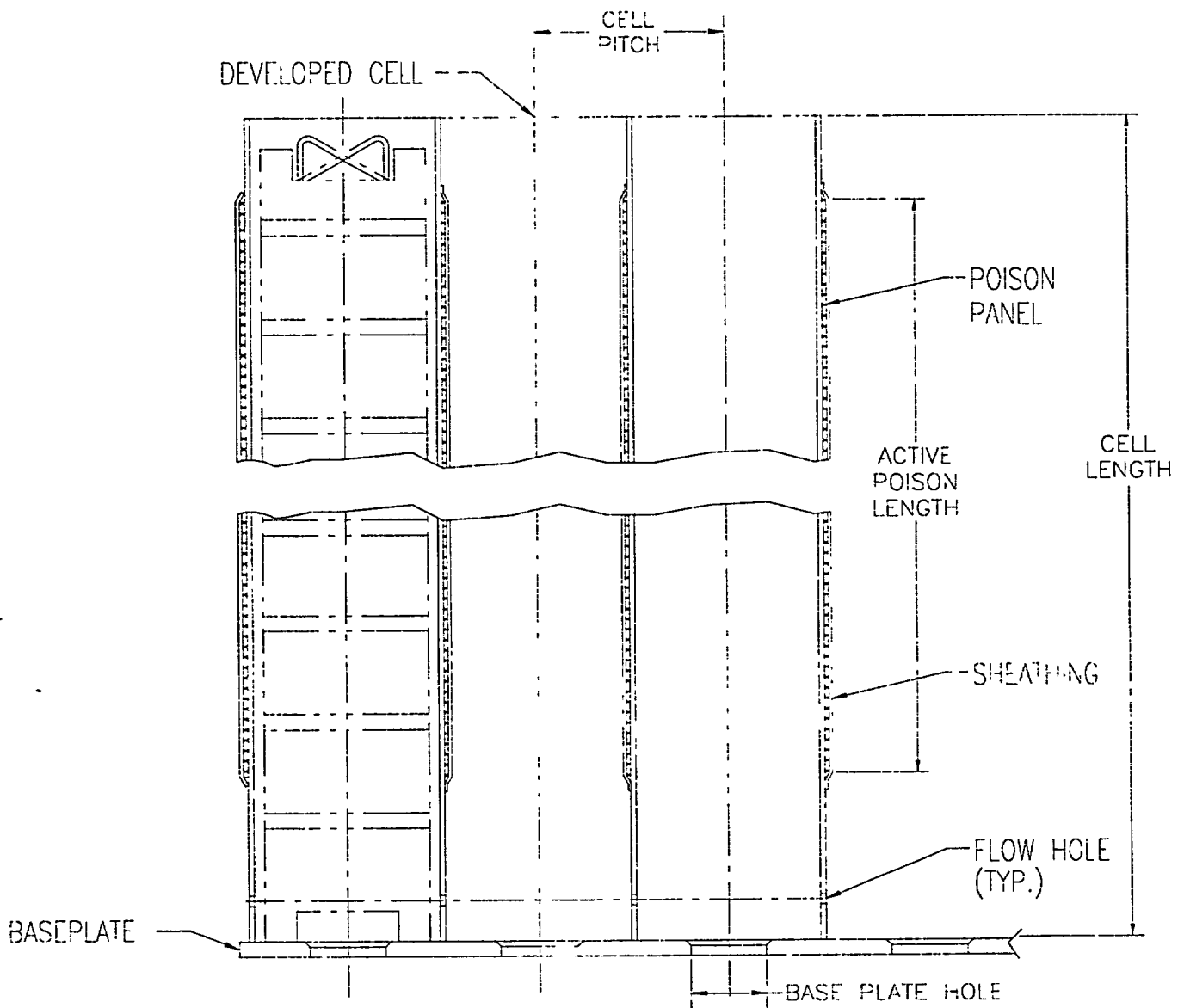


FIGURE 2.6.8; ELEVATION VIEW OF STORAGE RACK MODULE

NOTE: DEPICTION OF STORED FUEL ASSEMBLY IS NOT
INTENDED TO BE ACCURATE.

3.0 MATERIAL AND HEAVY LOAD CONSIDERATIONS

3.1 Introduction

Safe storage of nuclear fuel in the pool requires that the materials utilized in the rack fabrication be of proven durability and compatible with the pool water environment. This section provides a synopsis of the considerations with regard to long-term design service life of 60 years.

3.2 Structural Materials

The following structural materials are utilized in the fabrication of the fuel racks:

- a. ASTM A240-304L for all sheet metal stock and baseplate
- b. Internally threaded support legs: ASTM A240-304L
- c. Externally threaded support spindle: ASTM A564-630 precipitation hardened stainless steel (heat treated to 1100°F)
- d. Weld material - ASTM Type 308

3.3 Neutron Absorbing Material

In addition to the structural and non-structural stainless material, the racks employ BoralTM, a patented product of AAR Manufacturing, as the neutron absorber material. A brief description of Boral, and its pool experience list follows.

Boral is a thermal neutron poison material composed of boron carbide and 1100 alloy aluminum. Boron carbide is a compound having a high boron content in a physically stable and chemically inert form. The 1100 alloy aluminum is a lightweight metal with high tensile strength, which is protected from corrosion by a highly resistant oxide film. The two materials, boron carbide and aluminum, are chemically compatible and ideally suited for long-term use in the radiation, thermal and chemical environment of a spent fuel pool. Boral has been shown [3.3.1] to be superior to alternative materials previously used as neutron absorbers in storage racks.

Boral has been exclusively used in fuel rack applications in recent years. Its use in spent fuel pools as the neutron absorbing material can be attributed to its proven performance (over 150 pool years of experience) and the following unique characteristics:

- i. The content and placement of boron carbide provides a very high removal cross-section for thermal neutrons.
- ii. Boron carbide, in the form of fine particles, is homogeneously dispersed throughout the central layer of the Boral panels.
- iii. The boron carbide and aluminum materials in Boral do not degrade as a result of long-term exposure to radiation.
- iv. The neutron absorbing central layer of Boral is clad with permanently bonded surfaces of aluminum.
- v. Boral is stable, strong, durable, and corrosion resistant.

Holtec International's Q.A. program ensures that Boral is manufactured by AAR Manufacturing under the control and surveillance of a Quality Assurance/Quality Control Program that conforms to the requirements of 10CFR50 Appendix B, "Quality Assurance Criteria for Nuclear Power Plants".

As indicated in Tables 3.3.1 and 3.3.2, Boral has been licensed by the USNRC for use in numerous BWR and PWR spent fuel storage racks and has been extensively used in international nuclear installations.

3.3.1 Boral Material Characteristics

Aluminum: Aluminum is a silvery-white, ductile metallic element. The 1100 alloy aluminum is used extensively in heat exchangers, pressure and storage tanks, chemical equipment, reflectors and sheet metal work.

It has high resistance to corrosion in industrial and marine atmospheres. Aluminum has atomic number of 13, atomic weight of 26.98, specific gravity of 2.69 and valence of 3. The physical, mechanical and chemical properties of the 1100 alloy aluminum are listed in Tables 3.3.3 and 3.3.4.

The excellent corrosion resistance of the 1100 alloy aluminum is provided by the protective oxide film that quickly develops on its surface from exposure to the atmosphere or water. This film prevents the loss of metal from general corrosion or pitting corrosion.

Boron Carbide: The boron carbide contained in Boral is a fine granulated powder that conforms to ASTM C-750-80 nuclear grade Type III. The material conforms to the chemical composition and properties listed in Table 3.3.5.

References [3.3.2], [3.3.3], and [3.3.4] provide further discussion as to the suitability of these materials for use in spent fuel storage module applications.

3.4 Compatibility with Environment

All materials used in the construction of the Holtec racks have been determined to be compatible with the St. Lucie Spent Fuel Pool, and have an established history of in-pool usage. As evidenced in Tables 3.3.1 and 3.3.2, Boral has been successfully used in both PWR and BWR fuel pools. Austenitic stainless steel (304L) is a widely used stainless alloy in nuclear power plants.

3.5 Heavy Load Considerations for the Proposed Rack Installations

The main hook of the Spent Fuel Cask Handling Crane for each Unit will be used for lifting the new rack and platforms into the respective Fuel Handling Building. Safe handling of heavy loads by the Spent Fuel Cask Handling Cranes will be ensured by following the defense in depth approach guidelines of NUREG 0612:

- Defined safe load paths in accordance with approved procedures
- Supervision of heavy load lifts by designated individuals
- Crane operator training and qualification that satisfies the requirements of ANSI/ASME B30.2-1976 [3.5.1]

- Use of lifting devices (slings) that are selected, inspected and maintained in accordance with ANSI B30.9-1971 [3.5.2]
- Inspection, testing and maintenance of cranes in accordance with ANSI/ASME B30.2-1976
- Ensuring the design of the Fuel Cask Cranes meets the requirements of CMAA-70 [3.5.3] and ANSI/ASME B30.2-1976
- Reliability of special lifting devices by application of design safety margins, and periodic inspection and examinations using approved procedures

The salient features of the lifting devices and associated procedures are described as follows:

a. Safe Load Paths and Procedures

Safe load paths will be defined for moving the new rack into the Fuel Handling Building (FHB). The rack will be lifted by the main hook of the Spent Fuel Cask Handling Crane and enter the FHB through the L-shaped door above the cask pit designed for ingress and egress of spent fuel casks. Therefore, the rack will enter the building at a location directly above the area of placement and need not be carried over portions of the Spent Fuel Pool.

A staging area will be setup outside of the FHB as a laydown area for the new rack. The staging area location also will not require any heavy load to be lifted over the SFP or any safety-related equipment.

All phases of rack installation activities will be conducted in accordance with written procedures, which will be reviewed and approved by the owner.

b. Supervision of Lifts

Procedures used during the installation of the Cask Pit racks require supervision of heavy load lifts by a designated individual who is responsible for ensuring procedure compliance and safe lifting practices.

c. Crane Operator Training

All crew members involved in the use of the lifting and upending equipment will be given training by Holtec International using a videotape-aided instruction course which has been utilized in previous rack installation operations.

d. Lifting Devices Design and Reliability

The Spent Fuel Cask Handling Crane for each Unit is located outdoors, at the north end of its respective Fuel Handling Building, where it can access the L-shaped hatch, the adjacent laydown areas and the access road. The cranes, which are of the overhead bridge type, will be refurbished and upgraded to single failure proof capability before the rack installation commences. The rated capacities for each crane will also be increased to 150 tons (main hoist) and 25 tons (auxiliary hoist). Electrical interlocks and the physical design of the buildings prevent the cranes from carrying a load over the fuel storage area of the spent fuel pool. A temporary hoist with an appropriate capacity may be attached to the Cask Handling Crane hook to prevent submergence of the hook.

The following table determines the maximum lift weight during the installation of the new racks.

Item	Weight (lbs)
Rack	34,200 (max.)
Lift Rig	1,100
Rigging	500
Total Lift	35,800

It is clear, based on the heaviest rack weight to be lifted, that the heaviest load will be well below the 150 ton rating of the Spent Fuel Cask Handling Crane main hook. The hoist to be used in conjunction with the Cask Handling Crane will be selected to provide an adequate load capacity and comply with NUREG-0612.

Remotely engaging lift rigs, meeting all requirements of NUREG-0612, will be used to lift the new rack modules. The new rack lift rigs consist of four independently loaded traction rods in a lift configuration. The individual lift rods have a safety factor of greater than 10. If one of the rods break, the load will still be supported by at least two

rods, which will have a safety factor of more than 5 against ultimate strength. Therefore, the lift rigs comply with the duality feature called for in Section 5.1.6 (3) of NUREG 0612.

The lift rigs have the following attributes:

- The traction rod is designed to prevent loss of its engagement with the rig in the locked position. Moreover, the locked configuration can be directly verified from above the pool water without the aid of an underwater camera.
- The stress analysis of the rig is carried out and the primary stress limits postulated in ANSI N14.6 [3.5.4] are met.
- The rig is load tested with 300% of the maximum weight to be lifted. The test weight is maintained in the air for 10 minutes. All critical weld joints are liquid penetrant examined to establish the soundness of all critical joints.

e. Crane Maintenance

The Spent Fuel Cask Handling Cranes are maintained functional per the St. Lucie preventative maintenance procedures.

The proposed heavy loads compliance will be in accordance with the guidelines of NUREG-0612, which calls for measures to "provide an adequate defense-in-depth for handling of heavy loads near spent fuel...". The NUREG-0612 guidelines cite four major causes of load handling accidents, namely

- i. operator errors
- ii. rigging failure
- iii. lack of adequate inspection
- iv. inadequate procedures

The rack installation ensures maximum emphasis on mitigating the potential load drop accidents by implementing measures to eliminate shortcomings in all aspects of the operation including the four

aforementioned areas. A summary of the measures specifically planned to deal with the major causes is provided below.

Operator errors: As mentioned above, comprehensive training will be provided to the installation crew. All training shall be in compliance with ANSI B30.2.

Rigging failure: The lifting device designed for handling and installation of the new racks has redundancies in the lift legs and lift eyes such that there are four independent load members in the new rack lift rig, and three independent load members in the existing rack lifting rig. Failure of any one load bearing member would not lead to uncontrolled lowering of the load. The rig complies with all provisions of ANSI 14.6-1993, including compliance with the primary stress criteria, load testing at 300% of maximum lift load, and dye examination of critical welds.

The rig designs are similar to the rigs used in the initial racking or the rerack of numerous other plants, such as Hope Creek, Millstone Unit 1, Indian Point Unit Two, Ulchin II, Laguna Verde, J.A. FitzPatrick, and Three Mile Island Unit 1.

Lack of adequate inspection: The designer of the racks has developed a set of inspection points that have been proven to eliminate any incidence of rework or erroneous installation in numerous prior rerack projects. Surveys and measurements are performed on the storage racks prior to and subsequent to placement into the Cask Pit to ensure that the as-built dimensions and installed locations are acceptable. Measurements of the pool and floor elevations are also performed to determine actual pool configuration and to allow height adjustments of the pedestals prior to rack installation. These inspections minimize rack manipulation during placement into the pool.

Inadequate procedures: Procedures will be developed to address operations pertaining to the rack installation effort, including, but not limited to, mobilization, rack handling, upending, lifting, installation, verticality, alignment, dummy gage testing, site safety, and ALARA compliance. The procedures will be the successors of the procedures successfully implemented in previous projects.

Table 3.5.1 provides a synopsis of the requirements delineated in NUREG-0612, and its intended compliance.

3.6 References

- [3.3.1] "Nuclear Engineering International," July 1997 issue, pp 20-23.
- [3.3.2] "Spent Fuel Storage Module Corrosion Report," Brooks & Perkins Report 554, June 1, 1977.
- [3.3.3] "Suitability of Brooks & Perkins Spent Fuel Storage Module for Use in PWR Storage Pools," Brooks & Perkins Report 578, July 7, 1978.
- [3.3.4] "Boral Neutron Absorbing/Shielding Material - Product Performance Report," Brooks & Perkins Report 624, July 20, 1982.
- [3.5.1] ANSI/ASME B30.2, "Overhead and Gantry Cranes, (Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist)," American Society of Mechanical Engineers, 1976.
- [3.5.2] ANSI B30.9, "Safety Standards for Slings," 1971.
- [3.5.3] CMAA Specification 70, "Electrical Overhead Traveling Cranes," Crane Manufacturers Association of America, Inc., 2000.
- [3.5.4] ANSI N14.6-1993, Standard for Special Lifting Devices for Shipping Containers Weighing 10000 Pounds or more for Nuclear Materials," American National Standard Institute, Inc., 1978.

Table 3.3.1			
BORAL EXPERIENCE LIST - PWRs			
Plant	Utility	Docket No.	Mfg. Year
Maine Yankee	Maine Yankee Atomic Power	50-309	1977
Donald C. Cook	Indiana & Michigan Electric	50-315/316	1979
Sequoyah 1,2	Tennessee Valley Authority	50-327/328	1979
Salem 1,2	Public Service Electric & Gas	50-272/311	1980
Zion 1,2	Commonwealth Edison	50-295/304L	1980
Bellefonte 1, 2	Tennessee Valley Authority	50-438/439	1981
Yankee Rowe	Yankee Atomic Power	50-29	1964/1983
Gosgen	Kernkraftwerk Gosgen-Daniken AG (Switzerland)		1984
Koeberg 1,2	ESCOM (South Africa)		1985
Beznau 1,2	Nordostschweizerische Kraftwerke AG (Switzerland)		1985
12 various Plants	Electricite de France (France)	--	1986
Indian Point 3	NY Power Authority	50-286	1987
Byron 1,2	Commonwealth Edison	50-454/455	1988
Braidwood 1,2	Commonwealth Edison	50-456/457	1988
Yankee Rowe	Yankee Atomic Power	50-29	1988
Three Mile Island I	GPU Nuclear	50-289	1990
Sequoyah (rerack)	Tennessee Valley Authority	50-327	1992
Donald C. Cook (rerack)	American Electric Power	50-315/316	1992
Beaver Valley Unit 1	Duquesne Light Company	50-334	1993
Fort Calhoun	Omaha Public Power District	50-285	1993

Table 3.3.1			
BORAL EXPERIENCE LIST - PWRs			
Plant	Utility	Docket No.	Mfg. Year
Zion 1 & 2 (rerack)	Commonwealth Edison	50-295/304L	1993
Salem Units 1 & 2 (rerack)	Public Gas and Electric Company	50-272/311	1995
Ulchin Unit 1	Korea Electric Power Company (Korea)	--	1995
Haddam Neck	Connecticut Yankee Atomic Power Company	50-213	1996
Ulchin Unit 2	Korea Electric Power Company (Korea)	--	1996
Kori-4	Korea Electric Power Company (Korea)	--	1996
Yonggwang 1,2	Korea Electric Power Company (Korea)	--	1996
Sizewell B	Nuclear Electric, plc (United Kingdom)	--	1997
Angra 1	Furnas Centrais-Elétricas SA (Brazil)	--	1997
Waterford 3	Entergy Operations	50-382	1997
Callaway	Union Electric	50-483	1998
Millstone 3	Northeast Utilities	50-423	1998
Davis-Besse	First Energy	50-346	1999
Wolf Creek	Wolf Creek Nuclear Operating	50-482	1999
Harris Pool 'C'	Carolina Power & Light	50-401	1999
Yonggwang 5/6	Korea Electric Power Company (Korea)	--	2001
Kewaunee	Wisconsin Public Service	50-305	2001

Table 3.3.2			
BORAL EXPERIENCE LIST - BWRs			
Plant	Utility	Docket No.	Mfg. Year
Cooper	Nebraska Public Power	50-298	1979
J.A. FitzPatrick	NY Power Authority	50-333	1978
Duane Arnold	Iowa Electric Light & Power	50-331	1979
Browns Ferry 1,2,3	Tennessee Valley Authority	50-259/260/296	1980
Brunswick 1,2	Carolina Power & Light	50-324/325	1981
Clinton	Illinois Power	50-461/462	1981
Dresden 2,3	Commonwealth Edison	50-237/249	1981
E.I. Hatch 1,2	Georgia Power	50-321/366	1981
Hope Creek	Public Service Electric & Gas	50-354/355	1985
Humboldt Bay	Pacific Gas & Electric Company	50-133	1985
LaCrosse	Dairyland Power	50-409	1976
Limerick 1,2	Philadelphia Electric Company	50-352/353	1980
Monticello	Northern States Power	50-263	1978
Peachbottom 2,3	Philadelphia Electric	50-277/278	1980
Perry 1,2	Cleveland Electric Illuminating	50-440/441	1979
Pilgrim	Boston Edison Company	50-293	1978
Susquehanna 1,2	Pennsylvania Power & Light	50-387,388	1979
Vermont Yankee	Vermont Yankee Atomic Power	50-271	1978/1986
Hope Creek	Public Service Electric & Gas	50-354/355	1989
Harris Pool 'B' †	Carolina Power & Light	50-401	1991
Duane Arnold	Iowa Electric Light & Power	50-331	1993
Pilgrim	Boston Edison Company	50-293	1993

Table 3.3.2			
BORAL EXPERIENCE LIST - BWRs			
Plant	Utility	Docket No.	Mfg. Year
LaSalle 1	Commonwealth Edison	50-373	1992
Millstone Unit 1	Northeast Utilities	50-245	1989
James A. FitzPatrick	NY Power Authority	50-333	1990
Hope Creek	Public Service Electric & Gas Company	50-354	1991
Duane Arnold Energy Center	Iowa Electric Power Company	50-331	1994
Limerick Units 1,2	PECO Energy	50-352/50-353	1994
Harris Pool 'B' †	Carolina Power & Light Company	50-401	1996
Chinshan 1,2	Taiwan Power Company (Taiwan)	--	1986
Kuosheng 1,2	Taiwan Power Company (Taiwan)	--	1991
Laguna Verde 1,2	Comision Federal de Electricidad (Mexico)	--	1991
Harris Pool 'B' †	Carolina Power & Light Company	50-401	1996
James A. FitzPatrick	NY Power Authority	50-333	1998
Vermont Yankee	Vermont Yankee	50-271	1998
Plant Hatch	Southern Nuclear	50-321	1999
Harris Pool 'C' †	Carolina Power & Light Company	50-401	1999
Byron/Braidwood	Commonwealth Edison	50-401	1999
Enrico Fermi Unit 2	Detroit Edison	50-305	2000

† Fabricated racks for storage of spent fuel transhipped from Brunswick.

Table 3.3.3	
1100 ALLOY ALUMINUM PHYSICAL CHARACTERISTICS	
Density	0.098 lb/in ³ 2.713 g/cm ³
Melting Range	1190°F - 1215°F 643° - 657°C
Thermal Conductivity (77°F)	128 BTU/hr/ft ² /F/ft 0.53 cal/sec/cm ² /°C/cm
Coefficient of Thermal Expansion (68°F - 212°F)	13.1 x 10 ⁻⁶ in/in-°F 23.6 x 10 ⁻⁶ cm/cm-°C
Specific Heat (221°F)	0.22 BTU/lb/°F 0.23 cal/g/°C
Modulus of Elasticity	10 x 10 ⁶ psi
Tensile Strength (75°F)	13,000 psi (annealed) 18,000 psi (as rolled)
Yield Strength (75°F)	5,000 psi (annealed) 17,000 psi (as rolled)
Elongation (75°F)	35-45% (annealed) 9-20% (as rolled)
Hardness (Brinell)	23 (annealed) 32 (as rolled)
Annealing Temperature	650°F 343°C

Table 3.3.4 CHEMICAL COMPOSITION - ALUMINUM (1100 ALLOY)	
99.00% min.	Aluminum
1.00% max.	Silicone and Iron
0.05-0.20% max.	Copper
0.05% max.	Manganese
0.10% max.	Zinc
0.15% max.	Other


Table 3.3.5	
CHEMICAL COMPOSITION AND PHYSICAL PROPERTIES OF BORON CARBIDE	
CHEMICAL COMPOSITION (WEIGHT PERCENT)	
Total boron	70.0 min.
B ¹⁰ isotopic content in natural boron	18.0
Boric oxide	3.0 max.
Iron	2.0 max.
Total boron plus total carbon	94.0 min.
PHYSICAL PROPERTIES	
Chemical formula	B ₄ C
Boron content (weight percent)	78.28%
Carbon content (weight percent)	21.72%
Crystal structure	rhombohedral
Density	0.0907 lb/in ³ 2.51 g/cm ³
Melting Point	4442°F 2450°C
Boiling Point	6332°F 3500°C
Boral Loading (minimum grams B ¹⁰ per cm ²)	

Table 3.5.1 HEAVY LOAD HANDLING COMPLIANCE MATRIX (NUREG-0612)	
Criterion	Compliance
1. Are safe load paths defined for the movement of heavy loads to minimize the potential of impact, if dropped, on irradiated fuel?	Yes
2. Will procedures be developed to cover: identification of required equipment, inspection and acceptance criteria required before movement of load, steps and proper sequence for handling the load, defining the safe load paths, and special precautions?	Yes
3. Will crane operators be trained and qualified?	Yes
4. Will special lifting devices meet the guidelines of ANSI 14.6-1993?	Yes
5. Will non-custom lifting devices be installed and used in accordance with ANSI B30.20 [3.5.5], latest edition?	Yes
6. Will the cranes be inspected and tested prior to use in rack installation?	Yes
7. Does the crane meet the requirements of ANSI B30.2-1976 and CMMA-70?	Yes

4.0 Criticality Safety Analyses

The criticality analyses reported here include the new Cask Pit racks to be installed in both Unit 1 and Unit 2 of the St. Lucie Nuclear Power Plant.

4.1 Unit 1 Cask Pit Rack

4.1.1 Introduction and Summary

The purpose of this evaluation is to document the criticality safety of the new fuel storage rack to be installed in the Cask Pit adjacent to the spent fuel pool of the FPL St. Lucie Unit 1 Nuclear Plant. The high density Region 1 rack is designed to assure that the effective neutron multiplication factor (k_{eff}) is equal to or less than 0.95 with the rack fully loaded with most reactive fuel assemblies authorized to be stored and flooded with unborated water at the temperature within the operating range corresponding to the highest reactivity. The maximum calculated reactivity includes margins for uncertainty in reactivity calculations including mechanical tolerances. All independent uncertainties are statistically combined, such that the final k_{eff} will be equal to or less than 0.95 with a 95% probability at a 95% confidence level.

The analysis uses the MCNP4a Monte Carlo code developed by the Los Alamos National Laboratory as the primary methodology for the calculations. CASMO4 was used to determine the reactivity effects of manufacturing tolerances and, as necessary, to assess the effect of fuel burning. As permitted in the USNRC guidelines, parametric evaluations were performed for manufacturing tolerances and the associated reactivity uncertainties were combined statistically. All calculations were made using an explicit model of the fuel and storage cell geometry. Results of these calculations are then used to define the reference reactivity that assures safe storage of fuel assemblies in the Cask Pit rack.

Potential abnormal and accident conditions have also been considered in this study. The temperature and void coefficient of reactivity are negative and the maximum design reactivity occurs at 50 °F. No misloading event was evaluated for the Unit 1 Cask Pit rack.

The criticality analysis was performed assuming unborated water as the moderator. Although such a condition is not realistic in the cask pit, which shares borated water with the spent fuel pool, it bounds all possible boron dilution accidents and conforms to the requirements of 10CFR50.68. In practice, the presence of moderator soluble boron at the Technical Specification limit assures a significantly lower reactivity in the cask pit rack.

In summary, results of this analysis confirm that the Unit 1 Cask Pit fuel storage rack can safely accommodate fresh fuel with initial enrichments up to 4.50 ± 0.05 wt%, with assurance that the maximum reactivity, including calculational and manufacturing uncertainties, will not exceed 0.95, with 95% probability at the 95% confidence level.

4.1.2 Analysis and Criteria Assumptions

To assure that the true reactivity will always be less than the calculated reactivity, the following conservative analysis criteria and assumptions are used in the analysis of the Cask Pit rack.

1. An infinite radial array of storage cells was assumed.
2. Neutron absorption in minor structural members is neglected, i.e., spacer grids are analytically replaced by water.
3. Moderator is assumed to be un-borated water at a temperature that results in highest reactivity (50°F or 10° C)
4. No credit is taken for the presence of the Uranium-234 or Uranium-236 isotopes in the fuel.
5. The analyses used the most reactive fuel assemblies amongst CE 14x14 or Framatome 14x14 fuel.
6. Fuel assembly is centered in the cell.
7. The fuel assembly designs used in the evaluation do not contain any gadolinia and the results of the analysis yields a higher reactivity and therefore bounds any fuel with Gd_2O_3 in the fuel.

4.1.3 Acceptance Criteria

The primary acceptance criterion for analysis of the Cask Pit rack is that, under a hypothetical condition of 0 ppm soluble boron in the cask pit, the maximum k_{eff} shall be less than or equal to 0.95, including calculational uncertainties and effects of mechanical tolerances. Applicable codes, standards, and regulations, or pertinent sections thereof, include the following:

- General Design Criterion 62, Prevention of Criticality in Fuel Storage and Handling.
- Code of Federal Regulation 10CFR50.68, Criticality Accident Requirements
- USNRC letter of April 14, 1978, to all Power Reactor Licensees - OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications
- ANSI-8.17-1984, Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors.
- L. Kopp, "Guidance On The Regulatory Requirements For Criticality Analysis Of Fuel Storage At Light-Water Reactor Power Plants", USNRC Internal Memorandum, L. Kopp to Timothy Collins, August 19, 1998.

4.1.4 Design and Input Data

4.1.4.1 Fuel Assembly Design Specifications

Two different fuel assembly designs were considered in the analyses; the CE 14x14 lattice and the Framatome 14x14 lattice. Table 4.1.4.1 provides the design details for the fuel assemblies [1].

4.1.4.2 Fuel Storage Cells

The nominal Cask Pit rack storage cell used for the criticality analyses is shown in Figure 4.1.1. The cell is composed of each box face of an 8.58 inch square (inside dimension) stainless steel box that has a wall thickness of 0.075 inches with Boral absorber material mounted on the outside. The fuel assemblies are assumed to be centrally located in each storage cell on a nominal lattice spacing of 10.30 inches. This forms a water flux-trap between Boral absorber panels of adjacent cells of [REDACTED] inches. The Boral absorber has a thickness of [REDACTED] inches and a nominal B-10 areal density of [REDACTED] g/cm² ([REDACTED] g/cm² minimum). The outer stainless steel sheath is [REDACTED] inches thick.

4.1.5 Methodology

The primary criticality analyses were performed with the three-dimensional MCNP Monte Carlo code [2] developed by the Los Alamos National Laboratory. Benchmark calculations, presented in Appendix A, indicate a bias of 0.0009 ± 0.0011 (95%/95%) [3].

KENO5a (4), a 3-dimensional multi-group Monte Carlo code developed by the Oak Ridge National Laboratory, was used as an independent check and to determine the reactivity-effect of eccentric fuel assembly positioning. In these calculations, the 238-group SCALE cross-section library was used, together with the Norderm integral treatment for U-238 resonance shielding effects. Benchmark calculations (Appendix A) showed a calculational bias of 0.0030 ± 0.0012 .

CASMO4, a two-dimensional deterministic code [5] using transmission probabilities, was used to evaluate the small (differential) reactivity effects of manufacturing tolerances. Validity of the CASMO4 code was established by comparison with results of the MCNP calculations for comparable cases.

In the geometric model used in the calculations, each fuel rod and each fuel assembly were explicitly described. Reflecting boundary conditions effectively defined an infinite radial array of storage cells. In the axial direction, a 30-cm water reflector was used to conservatively describe axial neutron leakage. Each stainless steel box and water within the box was explicitly described in the calculational model.

Monte Carlo calculations inherently include a statistical uncertainty due to the random nature of neutron tracking. To minimize the statistical uncertainty of the calculated reactivities, a minimum of 3 million neutron histories was accumulated in each calculation.

4.1.6 Analysis Results

4.1.6.1 Reference Fuel Assembly

Table 4.1.4.1 summarizes the two fresh fuel assembly designs expected to be stored in the Region 1 Cask Pit rack. Calculations were made to confirm the most reactive fuel assembly of those listed in Table 4.1.4.1 and the results are summarized below.

Fuel Assembly	k_{inf} (CASMO @4.5 % Enrichment)
CE 14x14	0.8925
Framatome 14x14	0.8936

These data confirm that the Framatome 14x14 lattice fuel is more reactive and it is used in all the subsequent calculations.

4.1.6.2 Evaluation of Uncertainties

Calculations were made to determine the uncertainties in reactivity associated with manufacturing tolerances. Tolerances that would increase reactivity were calculated; negative values are expected to be of equal magnitude but opposite in sign over the small tolerance variations. Results of these calculations are shown in Table 4.1.6.1. The reactivity effects were separately evaluated in a sensitivity study for each independent tolerance and the results were combined statistically. Tolerances considered include the following:

4.1.6.2.1 Boron Loading Tolerance

The Boral absorber panels used in the storage cells are nominally [REDACTED] inch thick, [REDACTED] inch wide and 140-inch long, with a nominal B-10 areal density of [REDACTED] g/cm² ([REDACTED] g/cm² minimum). Differential CASMO-4 calculations indicate that the Boron loading tolerance limits result in incremental reactivity uncertainty shown in Table 4.1.6.1.

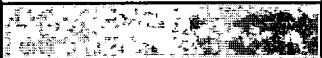
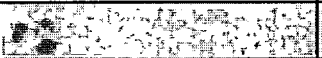


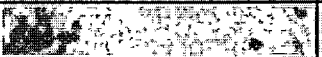


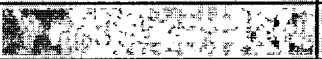
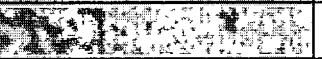

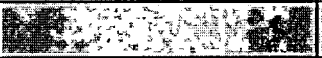
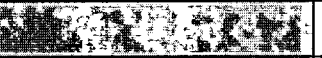


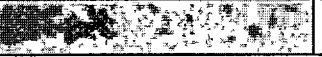
4.1.6.2.2 Boral Width Tolerance

The reference storage cell design uses a Boral panel with a width of 7.25 [REDACTED] inches. For the maximum tolerance, the calculated reactivity uncertainty is shown in Table 4.1.6.1 as determined by differential CASMO-4 calculations.

4.1.6.2.3 Tolerances in Water Gap Spacing, Cell Box Inner Dimension and Lattice Pitch


The design storage cell lattice spacing between fuel assemblies (10.3 [REDACTED] inches) results in a water-gap of 1.303 [REDACTED] inches. A decrease in lattice pitch or in water-gap (flux-trap) spacing or an increase in storage box I.D. increases reactivity. The inner stainless steel box dimension, 8.58 [REDACTED] inches, defines the storage box in which the fuel is stored. Tolerances on the three spacing dimensions are inter-related and all three tolerances are independently controlled. For example, the minimum lattice pitch tolerance of [REDACTED] inches can occur only for either (1) a decrease in water-gap thickness of [REDACTED] inches or (2) a decrease of [REDACTED] inches in box I.D. concurrent with a decrease of [REDACTED] inches in water gap

thickness. The bounding conditions for tolerances and the corresponding reactivity effect are listed below:


<u>TOLERANCES</u>			
Cell Box I.D.	Water Gap Thickness	Lattice Pitch	Reactivity Effect, Δk
			+0.0017
			-0.0012
			+0.0087
			+0.0035
			+0.0095

The last case above represents the largest reactivity effect of the dimensional tolerances. This uncertainty value, 0.0095 is used in Table 4.1.6.1.

4.1.6.2.4 Stainless Steel Thickness Tolerances

The nominal stainless steel thickness for the stainless steel box is 0.075 inches with a tolerance of  inches (standard ASME sheet metal tolerance). The maximum positive reactivity effect of the expected stainless steel box thickness tolerance is shown in Table 4.1.6.1.

4.1.6.2.5 Fuel Enrichment and Density Tolerances

The nominal U-235 design enrichment for this analysis is $4.50 \pm 0.05\%$. Evaluation for the maximum enrichment of 4.55 wt% yielded an incremental reactivity effect for the enrichment tolerance as shown in Table 4.1.6.1. Calculations were also made with the fuel density increased by 5% to the maximum expected value of  g/cm³ (stack density). Results are also given in Table 4.1.6.1 for the effect of this uncertainty in reactivity .

4.1.6.2.6 Sheathing Thickness

The stainless steel sheath is nominally [REDACTED] inches thick with a standard ASME sheet metal tolerance of [REDACTED] inches. For this tolerance, the calculated reactivity uncertainty is listed in Table 4.1.6.1.

4.1.6.2.7 Fuel Assembly Dimensional Tolerances

CASMO-4 calculations were made for various tolerances in the fuel assembly geometry. From these calculations, the incremental reactivity effects of each independent tolerance were determined. The tolerance effects calculated include the following:

- Tolerance in fuel rod pitch, statistically averaged for the 14x14 fuel rod array; $k=0.8984$; $\Delta k=0.0048$
- Tolerance in fuel pellet OD (\pm [REDACTED] inch); $k_{inf}=0.8941$; $\Delta k=0.0005$
- Tolerance in fuel clad thickness (maximum thickness used)
- Tolerance in Guide Tube OD (\pm [REDACTED] inches); $k_{inf}=0.8935$; $\Delta k=0.0001$
- Tolerance in Guide Tube Wall Thickness (\pm [REDACTED] inches); $k_{inf}=0.8941$; $\Delta k=0.0005$

The statistical sum of these reactivity tolerances is shown in Table 4.1.6.1 and is combined with the other tolerances.

4.1.6.3 Eccentric Positioning of Fuel Assembly

The fuel assembly is assumed to be normally located in the center of the storage rack cell. KENO5a calculations were also made with the fuel assemblies assumed to be in the corner of the storage rack cell (four eccentric assembly cluster at closest approach). These calculations indicated that the reactivity is slightly lower for the eccentric position and, therefore, the maximum reactivity occurs for the normal centered position of the fuel, as shown in Table 4.1.6.3.

4.1.6.4 Abnormal and Accident Conditions

4.1.6.4.1 Temperature and Void Effects

The moderator (bulk water) temperature coefficient of reactivity is negative; a minimum moderator temperature of 50 °F (10 °C) was assumed, which assures that the true reactivity will be lower for any value of water temperatures above 50 °F. Temperature effects on reactivity along with the effect of voids on reactivity are shown in Table 4.1.6.2. Introducing voids in the water (to simulate boiling) decreased reactivity, as shown in the table.

4.1.6.4.2 Dropped Fuel Assembly

For a drop on top of the rack, the fuel assembly will come to rest horizontally on top of the rack with a minimum separation distance from the fuel in the rack of more than 12 inches, including the effect of any deformation resulting from seismic or accident conditions. At this separation distance, the effect on reactivity is insignificant. Furthermore, soluble boron in the pool water would substantially reduce the reactivity and assure that the true reactivity is always less than the limiting value for any conceivable dropped fuel accident.

If the dropped fuel assembly were to enter a storage cell vertically and impact the base plate, the base plate could experience a local deformation estimated at 2 inches or less. This magnitude of deformation causes no significant changes to reactivity despite the fact that the dropped assembly would have a small amount of fuel exposed below the Boral absorber. This exposed fuel occurs in a high neutron leakage area and hence, the positive reactivity effect is minimal as shown in Table 4.1.6.3. Conservative calculations, assuming that the 2 inch deformation occurred everywhere on the base plate, showed a very small increase in reactivity (+0.0001 Δk).

4.1.6.4.3 Abnormal Location of a Fuel Assembly

The cask pit rack is designed for storage of fresh fuel assemblies with maximum initial enrichment of 4.50 ± 0.05 wt%. Hence, no internal fuel misloading accident is applicable. Due to the small clearance between the Cask Pit rack outer envelope and the Cask Pit liner, a fuel assembly cannot be positioned outside and adjacent to the cask pit rack. Thus, no accident scenario evaluation was performed.

4.1.6.5 Criticality Analyses Results

A summary of the results of the criticality safety analysis for the storage of fresh fuel (initial enrichment of 4.50 ± 0.05 wt%) is given in Table 4.1.6.4. The table also contains the calculational biases and the uncertainties. The results indicate that the maximum calculated reactivity in the new CPR will be 0.9061, and therefore storage of fresh fuel with initial enrichment up to 4.50 ± 0.05 wt% meets the regulatory requirements.

4.2 Unit 2 Cask Pit Rack

4.2.1 Objectives and General Description

The objective of the criticality safety analysis presented in this section is to document the requirements for safe storage of spent fuel assemblies in the Region 2 St. Lucie Unit 2 cask pit storage rack. This rack uses Boral as the poison material. The presence of Gadolinium poison in the fuel assembly lattice has been considered but not credited in the present analysis. Postulated accident conditions, where a fresh fuel assembly is inadvertently placed outside the rack or into a cell intended to contain a spent fuel assembly, have also been evaluated. The design criteria are such that no soluble boron is required in the pool water to protect against a mis-loaded assembly accident.

The analysis uses the MCNP4a Monte Carlo code developed by the Los Alamos National Laboratory as the primary code for the calculations. CASMO4 was used for calculation of spent fuel composition as well as to determine reactivity-effects of manufacturing tolerances. As permitted in the USNRC guidelines, parametric evaluations were performed for manufacturing tolerances and the associated reactivity uncertainties were combined statistically. All calculations were made for an explicit modeling of the fuel and storage cell geometries to define the enrichment-burnup combinations for spent fuel assemblies that assure a safe storage of spent fuel assemblies in the Region 2 cask pit rack.

The criticality analysis was performed assuming unborated water as the moderator. Although such a condition is not realistic in the cask pit, which shares borated water with the spent fuel pool, it bounds all possible boron dilution accidents and conforms to the requirements of 10CFR50.68. In practice, the presence of moderator soluble boron at the Technical Specification limit assures a significantly lower reactivity in the cask pit rack.

4.2.1.2 Summary of Results

The design specifications provide that the minimum burnup for the spent fuel (at an initial enrichment of 4.50 ± 0.05 wt%) in the cask pit rack, a Region 2 style configuration, is 36,000 MWD/MTU. A summary of the calculation results for spent fuel with initial enrichment of 4.50 ± 0.05 wt% is given in

Table 4.2.6.4. This table shows that the maximum k_{eff} under non-accident conditions is less than 0.916, which easily meets the acceptance criterion of $= 0.95$. The result for fuel assemblies with enrichments less than 4.50 ± 0.05 wt% is illustrated in Figure 4.2.1, where the maximum reactivities on the curve are all the same.

Evaluation of postulated accident conditions demonstrate that, for the most significant fuel assembly mis-loading accident, the maximum reactivity (k_{eff} of 0.9417), including bias and uncertainties, remains below 0.95 and no soluble boron is required to mitigate the effect of this postulated accident in the cask pit rack.

4.2.2 Analysis Criteria And Assumptions

To assure the true reactivity will always be less than the calculated reactivity, the following conservative analysis criteria or assumptions were used.

- Criticality safety analyses were based upon an infinite radial array of cells; i.e., no credit was taken for radial neutron leakage.
- Neutron absorption by minor structural materials was neglected; i.e., spacer grids were conservatively assumed to be replaced by water.
- Moderator is assumed to be un-borated water at a minimum temperature of 10°C (50°F).
- No axial blankets were assumed to be present in the fuel rods. The entire active fuel length was assumed to have the same enrichment.
- Credit for the reduction in reactivity with post-operation cooling time is not incorporated in the analysis.
- A conservative axial burnup distribution is used in evaluating the reactivity bias due to the burnup distribution (sometimes called the "end effect".)
- A 3-dimensional analysis with 10 axial zones is used in evaluating the end effect.
- The most reactive fuel assembly is assumed for the accident evaluation.

- The most reactive fuel assembly amongst the three fuel types to be stored in the Region 2 cask pit rack was used in the criticality safety analyses.
- Fuel assembly is centered in the cell.

4.2.3 Acceptance Criteria

The primary acceptance criteria, in accordance with 10CFR50.68, is that (1) the storage racks remain sub critical, under the postulated accident of the loss of all soluble boron, including bias and uncertainties, and (2) that partial credit for the soluble boron present may be taken to maintain the maximum k_{eff} less than or equal to 0.95. The maximum k_{eff} includes calculation bias and uncertainties as well as the reactivity effects of mechanical tolerances, and was evaluated under the postulated accident of the loss of all soluble boron.

Applicable codes, standards, and regulations, or pertinent sections thereof, include the following:

- Code of Federal Regulations, 10CFR50, General Design Criterion 62, Prevention of Criticality in Fuel Storage and Handling.
- Code of Federal Regulation 10CFR50.68, "Criticality Accident Requirements"
- USNRC Standard Review Plan, NUREG-0800, Section 9.1.2, Spent Fuel Storage.
- USNRC letter of April 14, 1978, to all Power Reactor Licensees - OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, including modification letter dated January 18, 1979.
- USNRC Regulatory Guide 1.13, Spent Fuel Storage Facility Design Basis, Rev. 2 (proposed), December, 1981.
- ANSI-8.17-1984, Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors.
- L. Kopp, "Guidance On The Regulatory Requirements For Criticality Analysis Of Fuel Storage At Light-Water Reactor Power Plants", USNRC Internal Memorandum L. Kopp to Timothy Collins, August 19, 1998.

4.2.4 Design And Input Data

4.2.4.1 Bounding Fuel Assembly

Calculations were made, using CASMO4, to evaluate the reactivity of the fuel assemblies currently in use or anticipated for storage in a St. Lucie Unit 2 Cask Pit Rack. Calculations in the cask pit rack, based on the fuel design parameters given in Table 4.2.4.1, show that the CE 16x16 fuel assembly exhibits the highest reactivity at the burnups of interest in this analysis and it was used in all the subsequent calculations.

Burnup, GWD/MTU	CE 14x14	Framatome 14x14	CE 16x16
10	1.0607	1.0594	1.0614
20	0.9916	0.9900	0.9930
30	0.9276	0.9254	0.9304
36	0.8892	0.8865	0.8930
40	0.8649	0.8618	0.8694

4.2.4.2 Storage Racks

A schematic of the cask pit fuel storage cell model, used in this analysis, is shown in Figure 4.2.2.

4.2.4.3 Operating Parameters

The principal core operating parameters, used in this study, are summarized in the table below.

PARAMETER	VALUE
Average Fuel Pellet Temperature	1604 °F
Hot Leg Moderator Temperature	606 °F
Average Core Soluble Boron Concentration	750 ppm

The reactivity effects of the gadolinia present in the fresh fuel assemblies have also been evaluated in this analysis. Based on 20 Gadolinia bearing rods of 8% Gd_2O_3 in each assembly, CASMO4 calculations were made with and without the gadolinia. These calculations are summarized below:

Burnup, GWD/MTU	k-inf with Gd_2O_3	k-inf without Gd_2O_3
1	0.9520	1.1295
10	0.9586	1.0614
20	0.9827	0.9930
30	0.9262	0.9304
36	0.8896	0.8930
45	0.8378	0.8406

Results of these calculations show that calculations without gadolinia are slightly more conservative (bounding) than with gadolinia present. Although gadolinia would be expected to harden the neutron spectrum (producing more plutonium), the poisoning effect of the residual gadolinia compensates for the higher plutonium production.

4.2.5 Methodology

The primary criticality analyses were performed with the three-dimensional MCNP Monte Carlo code [3] developed by the Los Alamos National Laboratory. Benchmark calculations, presented in Appendix A, indicate a bias of 0.0009 ± 0.0011 (95%/95%) [4].

CASMO4, a two-dimensional deterministic code [5] using transmission probabilities, was used to evaluate the small (differential) reactivity effects of manufacturing tolerances. Validity of the CASMO4 code was established by comparison with results of the MCNP calculations for comparable cases.

In the geometric model used in the calculations, each fuel rod and each fuel assembly were explicitly described. Reflecting boundary conditions effectively defined an infinite radial array of storage cells. In the axial direction, a 30-cm water reflector was used to conservatively describe axial neutron leakage. Each stainless steel box and water within the box was explicitly described in the calculational model.

Monte Carlo (MCNP) calculations inherently include a statistical uncertainty due to the random nature of neutron tracking. To minimize the statistical uncertainty of the calculated reactivities, a minimum of 3 million neutron histories was accumulated in each calculation. MCNP cannot perform depletion calculations, and depletion calculations were performed with CASMO4. Explicit description of the fission product nuclide concentrations in the spent fuel was determined from the CASMO4 calculations and used in the MCNP calculations. To compensate for those few fission product nuclides that cannot be described in MCNP, an equivalent amount of boron-10 in the fuel was determined which produced very nearly the same reactivity in MCNP as the CASMO4 result. This methodology explicitly incorporates approximately 40 of the most important fission products, accounting for all but about 1% in k . The remaining ~1 % in k is included by the equivalent B-10 concentration in the fuel.

4.2.6 Evaluation of Uncertainties

4.2.6.1 Uncertainty in Manufacturing Tolerances

CASMO4 calculations were made to determine the uncertainties in reactivity associated with tolerances in the rack's dimensions, fuel density and fuel enrichments. The reactivity effects of each independent tolerance were combined statistically. The rack dimensions and tolerances are shown in Figure 4.2.2.

For estimating the reactivity uncertainties associated with tolerances in fuel enrichment and density, tolerances of $\pm 0.05\%$ in enrichment and $\pm 1\%$ in UO_2 density were assumed. The reactivity associated with the fuel density tolerance is listed in Table 4.2.6.1. The reactivity effects of the tolerances in the rack dimensions are also listed in Table 4.2.6.1. The reactivity effects for the tolerance in fuel enrichment are listed in Table 4.2.6.2.

4.2.6.2 Uncertainty in Depletion Calculations

The uncertainty in depletion calculations is part of the methodology uncertainty and was taken as 5% of the reactivity decrement from beginning-of-life to the burnup of concern for the spent fuel [8]. This methodology uncertainty is included in the calculations of the final k_{eff} in Table 4.2.6.4.

4.2.6.3 Eccentric Location of Fuel Assemblies

The fuel assemblies are nominally stored in the center of the storage cells. Eccentric positioning of fuel assemblies in the cells normally results in a negligible effect or a reduction in reactivity for poisoned racks. Calculations have been made confirming negative reactivity effect of the eccentric positioning of four fuel assemblies at the position of closest approach. These calculations gave a small reduction in k_{eff} (-0.0013) confirming that eccentric positioning of fuel has a negligible effect.

4.2.6.4 Temperature and Void Effects

Temperature effects were also evaluated, using CASMO4, in the temperature range from 10 °C to 120 °C and the results are listed in Table 4.2.6.3. These results show that the temperature coefficient of reactivity is negative. The void coefficient of reactivity (boiling conditions) was also found to be negative for the St. Lucie Unit 2 cask pit rack. The reference temperature is 20 °C. The reactivity effects of pool water temperatures below 20 °C to 10 °C are calculated using CASMO (Table 4.2.6.3). These data were interpolated for water temperature to 10 °C (50 °F) and the resulting reactivity increment is added to the calculated k_{eff} at 20 °C.

4.2.6.5 Reactivity Effect of the Axial Burnup Distribution

Initially, fuel loaded into the reactor will burn with a slightly skewed cosine power distribution. As burnup progresses, the burnup distribution will tend to flatten, becoming more highly burned in the central regions than in the upper and lower ends. The more reactive fuel near the ends of the fuel assembly (less than average burnup) has reactivities slightly above that of the assembly average. Axial burnup penalty calculations based upon a conservative burnup distribution [6,9], gave a positive reactivity effect of the axial burnup distribution of $0.0071\Delta k$ for spent fuel of 36 MWD/KgU. These calculations are based on 10 zone axial calculations, using specific (CASMO) concentrations of actinides and fission products in each zone. Calculations for 4% fuel at 30 MWD/KgU gave a correction of $0.0007\Delta k$ and the correction becomes negative (neglected) below a burnup of ~ 29 MWD/KgU.

4.2.7 Accident Conditions And Soluble Boron Requirements

The accident scenarios considered in this analysis are summarized below:

- A dropped fuel assembly coming to rest horizontally across the top of the storage cell.
- A dropped fuel assembly, which enters the storage cell vertically and impacts the base plate.
- An extraneous assembly positioned outside and immediately adjacent to the storage rack

- The effects of a fresh fuel assembly mis-loaded into a cell intended to store a spent fuel assembly.

Among these, the most serious postulated accident condition is the misplacement of a fresh fuel assembly into a location intended for storage of a spent fuel assembly. Misplacement of a fuel assembly outside the periphery of a storage module is bounded by the more serious accident of a mis-placed assembly internal to the rack. This is due to the fact that the peripheral region between the rack and the wall is high neutron leakage area and Boral panels are present on the periphery of the cask pit rack. A dropped assembly lying on top of the rack would have a negligible reactivity effect because of the separation distance. If the dropped fuel assembly were to enter a storage cell vertically and impact the base plate, the base plate could experience a local deformation estimated at 2 inches or less. This magnitude of deformation causes no significant changes to reactivity despite the fact that the dropped assembly would have a small amount of fuel exposed below the Boral absorber. This exposed fuel occurs in a high neutron leakage area and hence, the positive reactivity effect is minimal as shown in Table 4.2.6.5. Conservative calculations, assuming that the 2 inch deformation occurred everywhere on the base plate, showed a negligible increase in reactivity (+0.0003 Δk).

The analysis shows that, for the most serious postulated accident condition with the internal misplacement of a fresh fuel assembly, the maximum reactivity (0.9417) remains well below 0.95 (including bias and uncertainties) and no soluble boron is required.

4.2.8 Criticality Analyses Results And Conclusions

Storage of spent fuel assemblies in the Unit 2 Region 2 cask pit storage rack has been evaluated in this analysis. The results are summarized in Tables 4.2.6.4 for fuel of 4.5% initial enrichment. Minimum burnup requirements for fuel of lower enrichments are shown in Figure 4.2.1 and listed in Table 4.2.8.1. These burnup requirements give reactivity values equivalent to those for the design basis case of 4.5% enriched fuel burned to 36 MWD/kgU. All points on the curve in Figure 4.2.1 have the same maximum reactivity and were evaluated in the same way as the design basis case, including appropriate bias and uncertainties. Temperature correction and MCNP4a bias were conservatively assumed to be independent of the initial enrichment. A summary of the conclusions are given below:

- The criticality margin of spent fuel assemblies (4.50 ± 0.05 wt% initial enrichment) stored in the cask pit rack of St. Lucie Unit 2 is acceptable within NRC guidelines and regulations. Storage of fuel assemblies of lower enrichment, conforming to the minimum burnup-enrichment combination shown in Figure 4.2.1, is permitted.
- Accident analysis show that soluble boron is not required to compensate for the reactivity effects of the most serious postulated fuel misplacement in the cask pit rack and the k_{eff} remains below 0.95, including all uncertainties and biases.

4.3 References

4.3.1 References for Section 4.1

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4.3.2 References For Section 4.2

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4. S. E. Turner, "Uncertainty Analysis – Burnup Distributions," Presented at the 1988 DOE/SANDIA Technical Meeting on Fuel Burnup Credit.
5. L. Kopp, "Guidance On The Regulatory Requirements For Criticality Analysis Of Fuel Storage At Light-Water Reactor Power Plants", USNRC Internal Memorandum, L. Kopp to Timothy Collins, August 19, 1998.

Table 4.1.4.1 Design Basis Fuel Assembly Specifications

PARAMETER	CE 14X14	Framatome 14X14
Rod Array Size	14x14	14x14
Rod Pitch (inches)	0.580±0.015	0.580±0.015
Active Fuel Length (inches)	136.7±0.50	136.7±0.50
Stack Density (gm/cm ³)	10.05 ± 5%	10.30 ± 5%
Total Number of Fueled Rods	176	176
Fuel Rod Outer Diameter (inches)	0.440±0.002	0.440±0.002
Cladding Thickness (inches)	0.026 - 0.028 ±0.002	0.028 - 0.031 ±0.003
Cladding Material	Zr-4	Zr-4
Pellet Diameter (inches)	0.3805±0.001	0.3770±0.001
Number of Guide/Instrument Tubes	5	5
Guide/Instrument Outer Diameter (inches)	1.115±0.003	1.115±0.003
Guide/Instrument Wall Thickness (inches)	0.040±0.004	0.040±0.004
Material	Zr-4	Zr-4

Table 4.1.6.1 Reactivity Effects of Manufacturing Tolerances in St. Lucie Unit 1 Nuclear Plant Cask Pit Storage Rack

PARAMETER	Value with Tolerance	k_{inf}	Δk
Reference k_{inf}	-	0.8936	-
Variation in Boral Panel B-10 Loading	0.028 g/cm ²	0.8958	±0.0022
Boral Panel Width	7.1875 inches	0.8945	±0.0009
Maximum Tolerance Effect of Box I.D., Water-gap Thickness, and Lattice Pitch (See Section 4.1.6.2.3)	-	0.9031	±0.0095
SS Box Wall Thickness	0.082 inches	0.8945	±0.0009
SS Sheathing Thickness	0.0238 inches	0.8934	±0.0002
Uncertainty in Maximum Fuel Enrichment	4.55 wt%	0.8954	±0.0018
Uncertainty in Fuel Density	10.815 gm/cm ³	0.8991	±0.0055
Fuel Assembly Dimensional Tolerance (Combined)	-	(See Section 4.1.6.2.7)	±0.0048
Statistical Sum	-	-	±0.0124

* All of the k_{inf} presented for tolerance effects are single variable effects.

Table 4.1.6.2. Reactivity Effects of Temperature and Void for CE 14x14 Fuel in St. Lucie Unit 1 Cask Pit Rack

T=10°C (50°F)		T = 20 °C		T = 60 °C		T = 120 °C		T = 120 °C + VOID	
k_{inf}	Δk^*	k_{inf}	Δk	k_{inf}	Δk^*	k_{inf}	Δk^*	k_{inf}	Δk
0.8945	+0.0009	0.8936	0	0.8872	-0.0064	0.8723	-0.0213	0.8448	-0.0488

* 20 °C is the reference temperature for calculations.

Table 4.1.6.3 Reactivity Effects of Abnormal And Accident Conditions (Unit 1 CPR)

Accident/Abnormal Conditions	Reactivity Effect
Temperature increase (above 50°F)	Negative
Void (boiling)	Negative (Table 6.2)
Assembly dropped on top of rack	Negligible ($<0.0001 \Delta k$)
Deep Drop of Fuel Assembly Through a Rack	Positive ($0.0001 \Delta k$)
Lateral rack module movement	NA
Mis-positioning of a fuel assembly within the Cask Pit Rack	NA
Eccentric Positioning of Fuel Assemblies	Negative

Table 4.1.6.4. Summary of the Criticality Safety Analyses for the Storage of Fresh Fuel Assemblies in St. Lucie Unit 1 Cask Pit Rack.








DESIGN BASIS ENRICHMENT	4.50±0.05 wt%
Reference k_{eff} (MCNP4a calculation)	0.8918
Calculational Bias, Δk	0.0009
Temperature (50°F)	0.0009
<u>Uncertainties</u>	
MCNP Bias Statistics (one sided tolerance factor [2] x standard deviation)	±0.0011
MCNP Statistics (1.7 x σ)	±0.0009
Manufacturing Tolerances	±0.0124
Eccentric Position	Negative
Statistical Combination of Uncertainties	±0.0125
Total	0.8936±0.0125
Maximum Reactivity (k_{inf})	0.9061

Table 4.2.4.1 Design Basis Fuel Assembly Specifications (Unit 2 CPR)

PARAMETER	VALUE		
	CE 14X14	Framatome 14X14	CE 16X16
Initial Enrichment, wt% U ²³⁵	4.50	4.50	4.50
Rod Array Size	14x14	14x14	16x16
Rod Pitch (inches)	0.580	0.580	0.506
Active Fuel Length (inches)	136.7	136.7	136.7
Uranium Stack Density*	96% of theoretical	96% of theoretical	96% of theoretical
Total Number of Fueled Rods	176	176	236
Fuel Rod Outer Diameter (inches)	0.440	0.440	0.382
Cladding Thickness (inches)	0.028	0.031	0.025
Cladding Material	Zr-4	Zr-4	Zr-4
Pellet Diameter (inches)	0.3805	0.3770	0.3255
Number of Guide/Instrument Tubes	5	5	5
Guide/Instrument Tube Outer Diameter (inches)	1.115	1.115	0.980
Guide/Instrument Tube Thickness (inches)	0.040	0.040	0.040
Material	Zr-4	Zr-4	Zr-4

* Fuel pellet dishing and chamfering not included.

Table 4.2.6.1 Reactivity Effects of Manufacturing Tolerances for CE 16x16 Fuel in St. Lucie Unit 2 Nuclear Plant Cask Pit Storage Rack.

Parameter, MWD/KgU	Value w/Tolerance	Burnup, 0 MWD/KgU		Burnup, 10 MWD/KgU		Burnup, 20 MWD/KgU		Burnup, 30 MWD/KgU		Burnup, 36 MWD/KgU	
		k_{inf}	Δk	k_{inf}	Δk	k_{inf}	Δk	k_{inf}	Δk	k_{inf}	Δk
Reference	-	1.1438	—	1.0614	—	0.9930	—	0.9304	—	0.8930	—
Minimum B-10	 g/sq- cm	1.1478	0.004	1.0652	0.0038	0.9965	0.0035	0.9337	0.0033	0.8962	0.0032
Boral Width	 cm	1.1448	0.001	1.0624	0.001	0.9939	0.0009	0.9312	0.0008	0.8938	0.0008
Min. Box ID	 in.	1.1465	0.0027	1.0639	0.0025	0.9952	.0022	0.9323	0.0019	0.8948	0.0018
SS Thickness	 in.	1.1448	0.001	1.0624	0.001	0.9938	0.0008	0.9311	0.0007	0.8937	0.0007
Guide Tube	 in.	1.1442	0.0004	—	—	—	—	—	—	—	—
Fuel Density	 g/cm ³	1.1446	0.0008	1.0621	0.0007	0.9938	0.0008	0.9314	0.001	0.8943	0.0013
Pellet OD	 in.	1.1442	0.0004	—	—	—	—	—	—	—	—
Statistical Sum*		0.0051		0.0048		0.0044		0.0041		0.0041	

* The statistical sum is the root-mean-square of the individual tolerance effects.

Table 4.2.6.2 Reactivity Effects of Fuel Enrichment Tolerance in St. Lucie Unit 2 Cask Pit Storage Rack

Burnup, MWD/KgU	B20	B20e	B25	B25e	B30	B30e	B35	B35e	B40	B40e	B45	B45e
1	0.92623	0.93314	0.98730	0.99256	1.03459	1.03874	1.07243	1.07581	1.10345	1.10626	1.12949	—
Δk	0.0069		0.0053		0.0042		0.0034		0.0028		—	
10	0.85689	0.86302	0.91345	0.91855	0.96044	0.96468	0.99975	1.00332	1.03298	1.03603	1.06144	1.06406
Δk	0.0061		0.0051		0.0042		0.0036		0.0031		0.0026	
20	0.78819	0.79353	0.83979	0.84469	0.88623	0.89056	0.92689	0.93064	0.96220	0.96547	0.99299	0.99585
Δk	0.0053		0.0049		0.0043		0.0038		0.0033		0.0029	
30	0.73391	0.73818	0.77750	0.78189	0.82065	0.82483	0.86102	0.86486	0.89766	0.90110	0.93036	0.93343
Δk	0.0043		0.0044		0.0042		0.0038		0.0034		0.0031	
36	0.70764	0.71113	0.74461	0.74850	0.78403	0.78798	0.82292	0.82670	0.85949	0.86299	0.89301	0.89617
Δk	0.0035		0.0039		0.0040		0.0038		0.0035		0.0032	

Table 4.2.6.3 Reactivity Effects of Temperature and Void in St. Lucie Unit 2 Cask Pit Storage Rack.

BURNUP, GWD/MTU	T = 20 °C	T = 10 °C	T=40°C	T = 80 °C	T=100°C	T = 120 °C	T = 120 °C + VOID
	k_{inf}	k_{inf}	k_{inf}	k_{inf}	k_{inf}	k_{inf}	k_{inf}
0	1.1438	1.1451	1.1404	1.1320	1.1269	1.1213	1.0979
10	1.0614	1.0627	1.0582	1.0501	1.0453	1.0400	1.0172
20	0.9930	0.9941	0.9900	0.9824	0.9780	0.9731	0.9506
30	0.9304	0.9313	0.9277	0.9209	0.9170	0.9126	0.8909
36	0.8930	0.8939	0.8906	0.8844	0.8807	0.8767	0.8555

Table 4.2.6.4 Summary of the Criticality Safety Analyses for the Storage of Spent Fuel Assemblies in the St. Lucie Unit 2 Cask Pit Rack.

Required Burnup of the Spent Fuel Assemblies	36 GWD/MTU
Initial Enrichment of Spent Fuel Assembly	4.5
MCNP calculated k_{eff} ³	0.8929 (0.9192 ⁽⁴⁾)
MCNP4a Bias	0.0009
Temperature Correction to 10 °C (50 °F)	0.0009
Axial Burnup Distribution Penalty	0.0071
Uncertainties	
MCNP4a Bias Uncertainty	± 0.0011
MCNP4a Statistics (95/95) Uncertainty ⁽¹⁾	± 0.0007
Manufacturing Tolerance Uncertainty	± 0.0052 ⁽²⁾
Depletion Uncertainty (5% of 1.1438-0.8930)	± 0.0125
Fuel Eccentric Positioning Uncertainty	Negative
Statistical Combination of Uncertainties	± 0.0136
Nominal k-eff	0.9014±0.0136
Maximum k_{eff}	0.9154 ⁽³⁾ (0.9417 ⁽⁴⁾)
Regulatory Limiting k_{eff}	0.9500

(1) $1.7 * \sigma$ ($\sigma = 0.0004$ or less)

(2) Statistical combination of tolerances from Table 4.2.6.1 and 4.2.6.2

(3) Maximum $k_{eff} = 0.9155$ by CASMO calculations.

(4) For the postulated fuel assembly mis-loaded accident.

Table 4.2.6.5 Reactivity Effects of Abnormal and Accident Conditions in St. Lucie Unit 2 Cask Pit Rack.

<u>ACCIDENT/ABNORMAL CONDITIONS</u>	<u>REACTIVITY EFFECT</u>
Temperature increase (See Table 6.3)	Negative
Void (Boiling) (See Table 6.3)	Negative
Misplacement of a fresh fuel assembly	Positive: for the most serious misplacement accident the maximum reactivity remains below 0.95
Deep Drop of Fuel Assembly Through a Rack	Positive (0.0003 Δk)
Eccentric Positioning of Fuel Assemblies	Negative

Table 4.2.8.1 Enrichment – Minimum Burnup Correlation (Ref. Figure 4.2.1). (Unit 2 CPR)

Initial Enrichment wt% U235	Minimum Required Burnup, MWD/KgU
2.0	5.80 (5.99)
2.5	11.80 (11.81)
3.0	17.61 (17.71)
3.5	23.67 (23.72)
4.0	29.47 (29.81)
4.5	36.00 (36.00)

Note: Values in the parenthesis are derived from the polynomial fit in Figure 1.

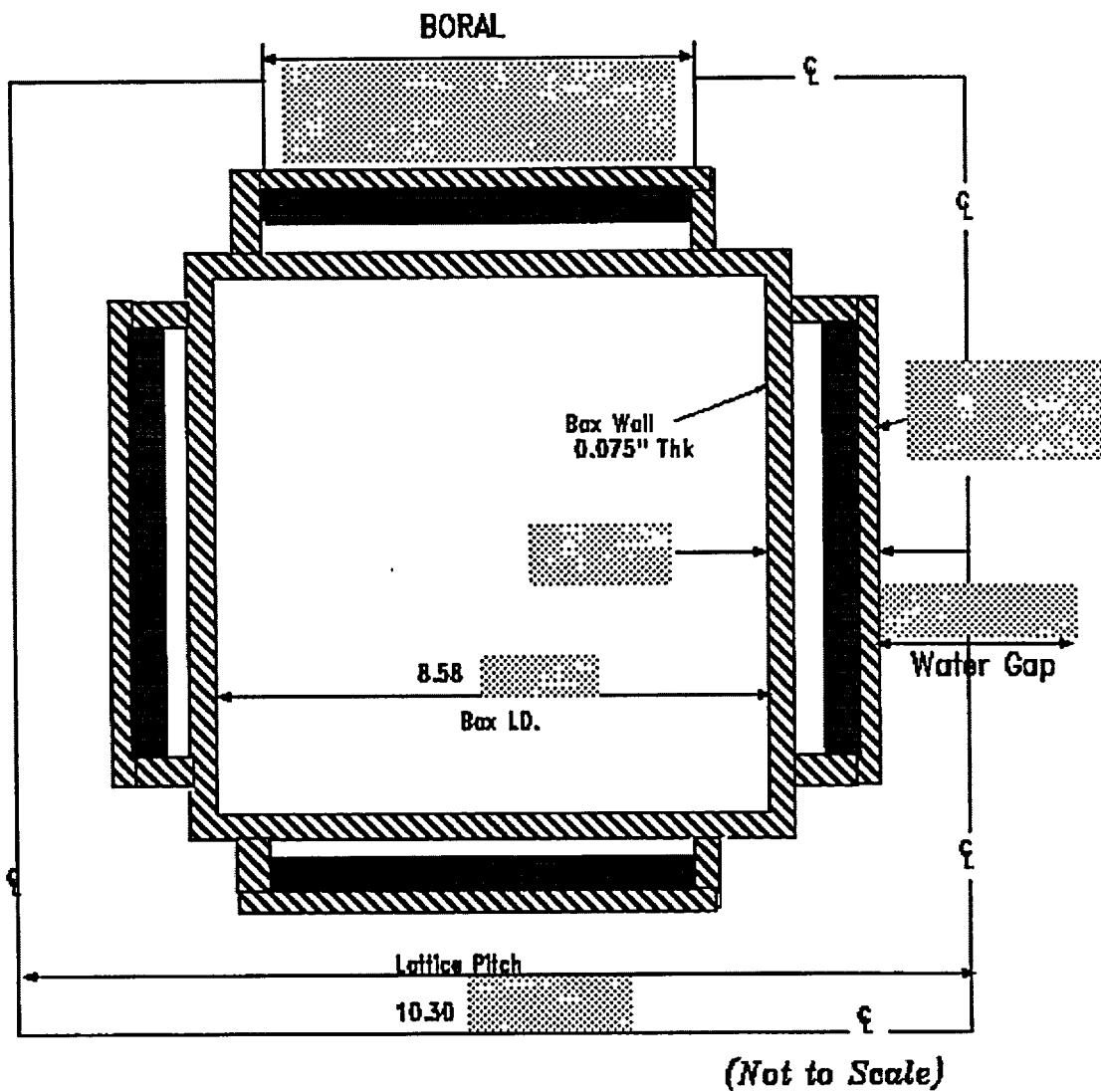


Figure 4.1.1; Unit 1 CPR Storage Cell Cross Section

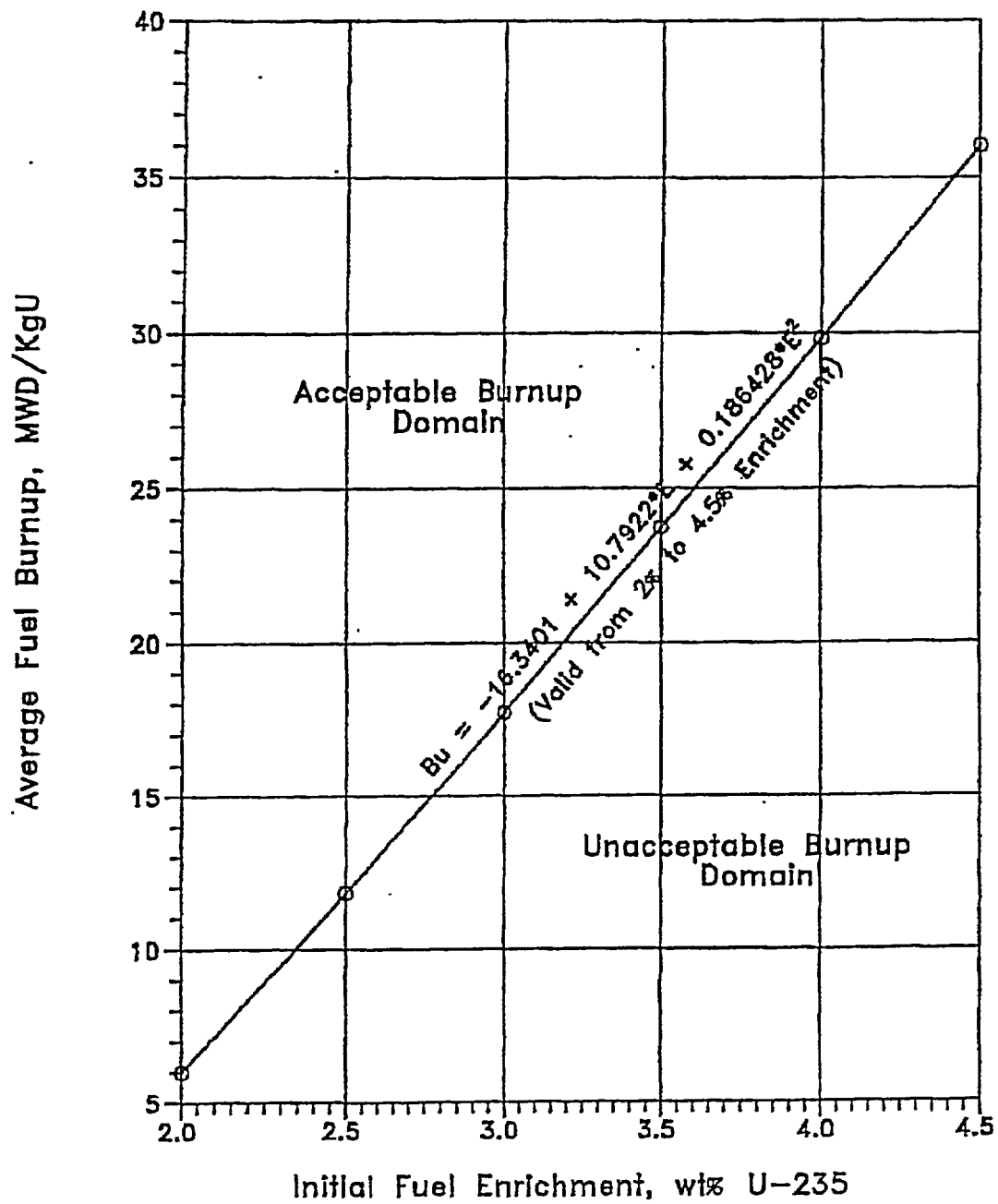


Figure 4.2.1; Limiting Fuel Burnup-Enrichment Combinations

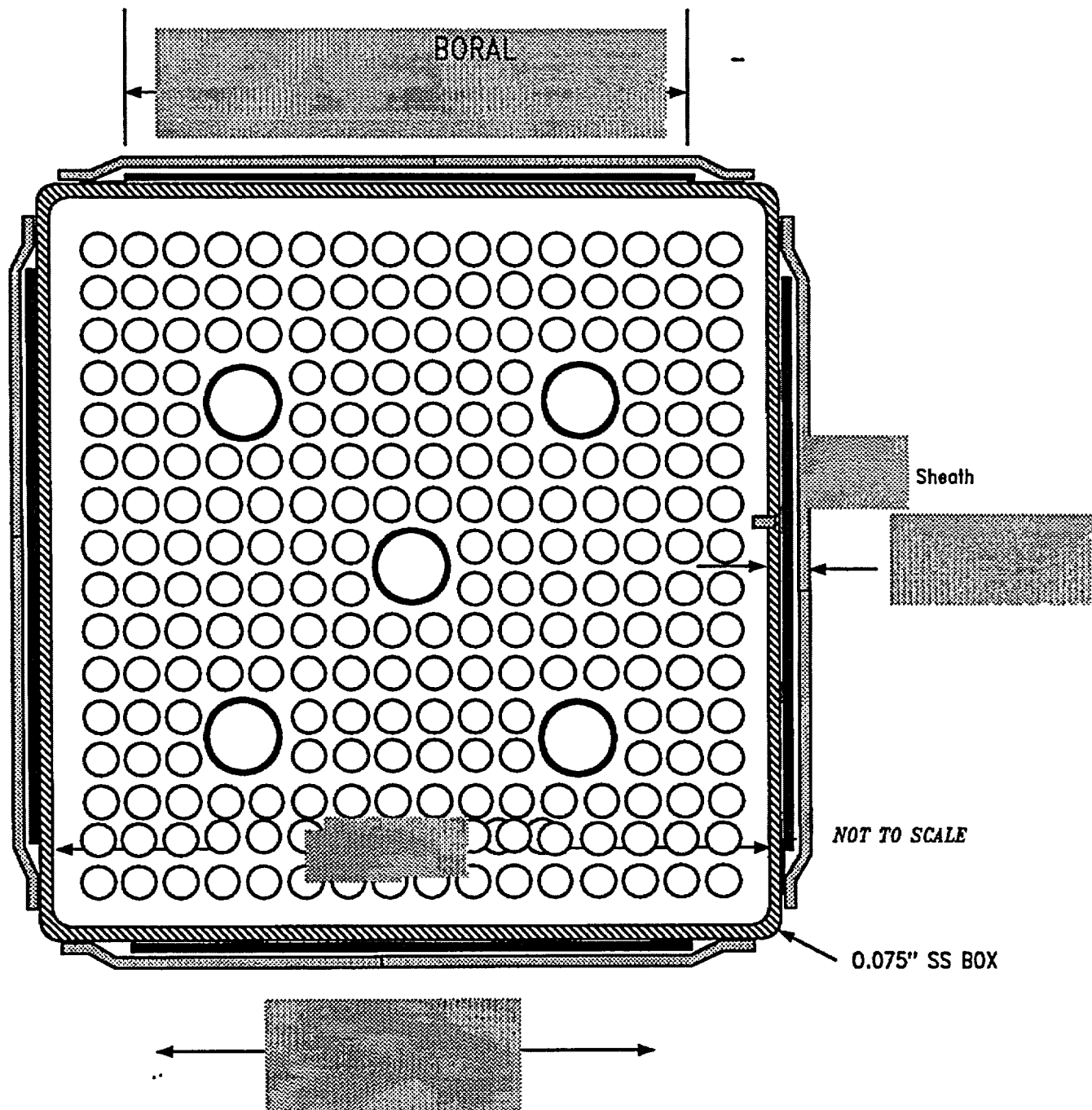


Figure 4.2.2 Unit 2 CPR STORAGE CELL Cross Section